
2. PROPOSED ACTION AND ALTERNATIVES

Chapter 2 describes the proposed action and alternatives to treat and manage the U.S. Department of Energy's inventory of sodium-bonded spent nuclear fuel and the No Action Alternative. The chapter includes discussions on the characteristics and quantities of the sodium-bonded spent nuclear fuel under consideration, the proposed treatment methods, and the potential sites and facilities for treatment or storage. It discusses why certain alternatives were dismissed from consideration. It also addresses issues associated with the ultimate disposition of the spent nuclear fuel and provides a summary comparison of the environmental impacts associated with the proposed action and the No Action Alternative.

2.1 INTRODUCTION

To fulfill the purpose and need discussed in Section 1.2, the U.S. Department of Energy (DOE) proposes to treat and manage sodium-bonded spent nuclear fuel and facilitate ultimate disposal in a geologic repository. The reasonable alternatives are determined by the technology options available to DOE to treat and manage the sodium-bonded spent nuclear fuel. To assist the reader in understanding the proposed action and reasonable alternatives, the following sections provide background information on the characteristics, inventory, and current storage locations of the sodium-bonded spent nuclear fuel; a discussion on the Electrometallurgical Treatment Research and Demonstration Project; the technology alternatives for its treatment and management; and the locations where these technologies could be implemented. The reasonable alternatives are discussed in Section 2.5.

2.2 SODIUM-BONDED SPENT NUCLEAR FUEL

As a result of research, development, and demonstration activities associated with liquid metal fast breeder reactors, DOE has approximately 60 metric tons of heavy metal of sodium-bonded spent nuclear fuel in its inventory. This represents approximately 2 percent of DOE's total current spent nuclear fuel inventory of nearly 2,500 metric tons of heavy metal. The common characteristic of sodium-bonded spent nuclear fuel is the presence of metallic sodium in the space between the cladding and the fuel and/or within the mass of the fuel. The presence of this chemically reactive material necessitates DOE's consideration of suitable treatment and management alternatives for this spent nuclear fuel before disposal in a geologic repository. Detailed descriptions of the characteristics of the sodium-bonded spent nuclear fuel in DOE's inventory are included in Appendix D.

Proposed Action

DOE proposes to treat and manage the sodium-bonded spent nuclear fuel in a safe and efficient manner and facilitate disposal in a geologic repository.

The bulk of sodium-bonded spent nuclear fuel in DOE's inventory is of two general types: driver fuel and blanket fuel. Driver fuel is used mainly in the center of the reactor core to "drive" and sustain the fission chain reaction. It is highly enriched in the fissile isotope uranium-235.

Blanket fuel is made from depleted uranium, a type of uranium in which most of the fissile uranium-235 has been removed. Blanket fuel usually is placed at the perimeter of the core and is used to breed the fissile material plutonium-239. It contains primarily the nonfissile isotope uranium-238, which converts to fissile plutonium-239 with the absorption of neutrons produced from the fission process. In some cases, as in the case of the Experimental Breeder Reactor-II (EBR-II), blanket fuel also has been used at the perimeter of the core for shielding. Typically, the fuel matrix in the sodium-bonded spent nuclear fuel is a uranium alloy or

uranium metal. A very small quantity (approximately 0.1 percent in mass of heavy metal) is in the form of uranium oxide, uranium or plutonium nitride, and uranium or plutonium carbide. Typical driver and blanket spent nuclear fuel elements are shown schematically in **Figure 2-1**.

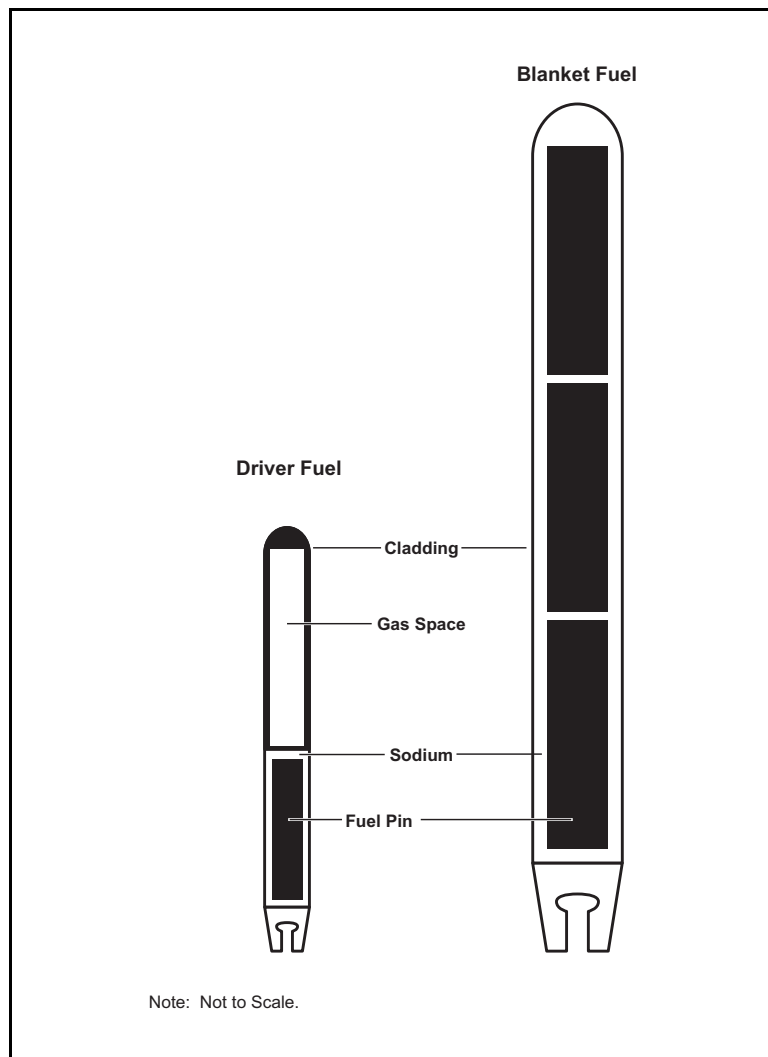


Figure 2-1 Typical Driver and Blanket Spent Nuclear Fuel Elements

mechanical stripping of the driver fuel cladding is not practical. The gap between the cladding and the fuel pin that contains sodium early in the irradiation lifetime disappears gradually due to outward swelling of the fuel pin. After approximately a 1 to 3 percent burnup, this gap is closed by swelling of the fuel pin due to irradiation effects such as fission gas bubble growth. Once contact between the fuel pin outer surface and cladding inner surface is made, cladding constituents (mainly nickel, chromium, and iron) gradually interdiffuse with fuel constituents (mainly uranium, plutonium, and zirconium) and the rare earth fission products (neodymium, cerium, lanthanum, praseodymium, samarium, and promethium) in the fuel. A solid-state layer bonding the fuel and cladding together is formed. This interdiffused layer effectively attaches the cladding to the fuel pin permanently in localized regions. Mechanical forces applied to these regions in a decladding operation either would leave pieces of the fuel pin attached to the cladding or vice versa. The resulting mix of cladding and fuel still would contain sodium inaccessible to subsequent treatment.

The blanket and driver spent nuclear fuel addressed in this environmental impact statement (EIS) contain metallic sodium between the cladding and the metallic fuel pins to improve the heat transfer from the fuel to the reactor coolant through the stainless steel cladding. When driver fuel is irradiated in the reactor for some period of time, the metallic fuel swells as fission products are generated until it reaches the cladding wall. Pores form throughout the fuel as it swells under pressure from the gaseous fission products. As these pores expand and connect to one another, the fission gases escape to a plenum in the fuel element just above the metallic fuel. As the gas escapes, the liquid sodium flows into these tiny pores, much like a sponge. As more pores form and grow, others are closed off from the fuel surface, including those containing sodium. Between 20 and 40 percent of the available sodium (up to 0.8 grams) may enter the driver fuel and become inseparable from the uranium except by dissolving or melting the fuel (Hofman and Walters 1994).

It also is well documented (Hofman and Walters 1994) that fuel and cladding components interdiffuse during irradiation to such an extent that

Blanket fuel, on the other hand, is at such a low burnup that significant swelling of the fuel pin does not occur. Gaps between the fuel pin and the cladding still exist at low burnup and little or no interdiffusion takes place. Cladding therefore can be mechanically removed from blanket fuel. Also, the swelling of this fuel is still low enough at typical blanket burnup that very little interconnected porosity exists. Hence, minimal sodium trapping would take place. Therefore, blanket fuel could be declad to effect sodium removal. The sodium removed from the sodium-bonded blanket spent nuclear fuel could be distilled and stabilized for disposal as low-level radioactive waste. The cladding would be disposed of as low-level radioactive waste or as part of the waste form being qualified for the proposed geologic repository.

2.2.1 EBR-II Spent Nuclear Fuel

EBR-II was a research and test reactor at Argonne National Laboratory-West (ANL-W) used to demonstrate the engineering feasibility of a sodium-cooled, liquid metal reactor with a steam electric power plant and integral fuel cycle. It achieved initial criticality in September 1961 and continued to operate until September 1994. During the 33 years of operation, numerous fuel designs were tested in EBR-II. EBR-II spent nuclear fuel contains both driver and blanket fuel.

The EBR-II driver spent nuclear fuel is stainless steel-clad, highly enriched uranium in a uranium alloy, typically either zirconium or fissium (an alloy of molybdenum, ruthenium, rhodium, palladium, zirconium, and niobium). There are some variations in the specific cladding alloys, enrichments, fuel compound alloys, dimensions, and burnup. When the fuel is “spent,” the enrichment (ratio of uranium-235 to total uranium) ranges between 55 and 76 percent. Each driver spent nuclear fuel element has a metal fuel pin about 36 centimeters (14 inches) long and less than 0.5 centimeters (0.2 inches) in diameter. The typical EBR-II driver spent nuclear fuel pin is a metal alloy of either 90 percent uranium and 10 percent zirconium or 95 percent uranium and 5 percent fissium. This fuel pin and a small amount of metallic sodium are loaded into a 74-centimeter-long (29-inch-long) stainless steel tube (cladding) and welded shut. This unit of fuel is called an element. Sixty-one (91 for some fuel) fuel elements are put together in a stainless steel hexagonal duct to make a fuel assembly approximately 2.3 meters (92 inches) long and 5.8 centimeters (2.3 inches) across. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Appendix D.

The EBR-II blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in metallic form. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary differences between the blanket designs are the dimensions. In EBR-II, the blanket assemblies were used primarily for shielding and for reducing the required size of the reactor core. Blanket assemblies were placed outside of a stainless steel shield for all but the first few years of EBR-II operation. Blanket assemblies are similar to driver assemblies, except that the individual blanket pins are larger. The blanket pins, made entirely from depleted uranium, are 1.1 centimeters (0.4 inches) in diameter. Three to five pins placed end-to-end make a sodium-bonded blanket element between 84 and 140 centimeters (33 to 55 inches) long. Since the blanket pins have a larger diameter, 19 blanket elements comprise a blanket assembly. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Appendix D.

The fuel from the last seven years of EBR-II operation is presently stored in three different locations at ANL-W (the Fuel Conditioning Facility, the Hot Fuel Examination Facility, and the Radioactive Scrap and Waste Facility) and two different locations at the Idaho Nuclear Technology and Engineering Center (INTEC), formerly the Idaho Chemical Processing Plant (ICPP). Previously, spent nuclear fuel was shipped to INTEC for reprocessing. However, INTEC ceased accepting the fuel in 1991 when a new uranium-zirconium alloy fuel, which could not be dissolved with INTEC’s existing plutonium-uranium extraction (PUREX) reprocessing system, went into full use at EBR-II. Prior to that, approximately 6 metric tons of EBR-II fuel were processed at INTEC. When DOE stopped processing at INTEC in 1992, elements from some 500 EBR-II driver spent

nuclear fuel assemblies of earlier design were left in storage pools (CPP-603 and CPP-666) located at INTEC. Water has been observed leaking into some of the storage containers in the CPP-603 storage pool, and the EBR-II fuel inside has reacted with the water and produced hydrogen gas. This is one of the reasons DOE is planning to remove all the spent nuclear fuel from the CPP-603 storage pool and place it in dry storage. National Environmental Policy Act (NEPA) coverage for this activity is provided by the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (Programmatic Spent Nuclear Fuel EIS) (DOE 1995a), and is not within the scope of this EIS.

2.2.2 Fermi-1 Spent Nuclear Fuel

Fermi-1 was designed and built at Monroe, Michigan (30 miles southwest of Detroit), to demonstrate the feasibility of the sodium-cooled, liquid metal fast breeder reactor for electric power production. Fermi-1 was a sodium-cooled, fast reactor. The reactor achieved initial criticality in 1963 and operated until September 1972. Fermi-1 was licensed for operation at a power level of 200 megawatts-thermal. Only blanket fuel from Fermi-1 is sodium-bonded.

The Fermi-1 blanket spent nuclear fuel consists of stainless steel-clad, depleted uranium in a uranium-molybdenum alloy. There are various blanket designs: upper and lower axial, and inner and outer radial blankets. The primary differences between these blanket designs are dimensions, elements per assembly, and burnup. Fermi-1 blanket elements are similar to EBR-II blanket elements in enrichment, but differ in dimensions (Fermi-1 elements are larger), form (uranium-molybdenum alloy versus uranium metal), and burnup. Because of its lower burnup, the Fermi-1 blanket fuel, which contains only about 0.2 percent plutonium by weight compared to approximately 1 percent plutonium by weight for the EBR-II blanket fuel, is subject to less stringent safeguard and security requirements than the EBR-II blanket fuel. This is an important consideration in the cost of managing the storage of these two types of fuel.

After the Fermi-1 reactor was permanently shut down, the blanket assemblies were placed into 14 canisters and transported to INTEC in 1974 and 1975 in 14 shipments. The canisters are made of stainless steel with a carbon steel basket inside. The canisters are 3.46 meters (134 inches) long and 64.8 centimeters (25.5 inches) in diameter. Twelve of the canisters contain the radial blanket subassemblies and 2 of the canisters contain the shorter axial blanket subassemblies. A subassembly is a cut assembly containing the blanket fuel pins. The canisters were placed into CPP-749, which is an underground dry storage system. The 14 canisters are stored in a single row of vaults on 4.6-meter (15-foot) centers.

The total quantity of Fermi-1 blanket material, both axial and radial, is 34 metric tons of heavy metal. The blanket assemblies have a very low irradiation history. Therefore, the inventory of fission products, activation products, and transuranics is low. The principal isotopes contributing to the activity of the axial and radial blanket assemblies are given in Appendix D.

2.2.3 Fast Flux Test Facility and Miscellaneous Sodium-Bonded Spent Nuclear Fuel

DOE's inventory of sodium-bonded spent nuclear fuel includes eight liquid metal reactor test assemblies containing driver spent nuclear fuel that were irradiated at the Fast Flux Test Facility at Hanford, Washington. It also includes small quantities of fuel that have metallic sodium or the alloy of sodium and potassium from liquid metal reactor experiments. These miscellaneous small-lot fuels differ in cladding composition, uranium content, enrichment, and burnup. Some of the fuel consists of uranium and/or plutonium carbides, nitrides, and oxides in addition to metal uranium or alloy. For the purposes of this EIS, this miscellaneous fuel is assumed to have driver fuel characteristics. This fuel is located at several DOE sites such as the Hanford site, Oak Ridge National Laboratory, the Savannah River Site (SRS), Sandia National Laboratories/New Mexico, and the Idaho National

Engineering and Environmental Laboratory (INEEL). Those lots stored outside INEEL will be transported to INEEL pursuant to the amended Record of Decision (61 FR 9441) for the Programmatic Spent Nuclear Fuel EIS (DOE 1995a). Under the proposed action, it is assumed that this spent fuel will be stored at INEEL.

Table 2–1 provides a summary of the spent nuclear fuel addressed by this EIS. As described earlier, the majority of the spent nuclear fuel consists of EBR-II driver fuel, EBR-II blanket fuel, and Fermi-1 blanket fuel.

Table 2–1 Overview of Sodium-Bonded Spent Nuclear Fuel Categories

<i>Spent Nuclear Fuel Type</i>	<i>Storage Volume ^a (cubic meters)</i>	<i>Metric Tons of Heavy Metal</i>	<i>Sodium Content (kilograms)</i>
EBR-II driver	58 ^b	3	83
EBR-II blanket	13	22	173
Fermi-1 blanket	19	34	365
Fast Flux Test Facility driver	8 ^b	0.3	7
Miscellaneous ^c	3 ^b	0.1	31
Total	101	60	662

^a Volume refers to the canister storage volume.

^b A larger volume per unit mass for the driver spent nuclear fuel is required for criticality control.

^c Assumed to have driver fuel characteristics.

Table 2–2 provides the site where the sodium-bonded spent nuclear fuel is stored, the locations within the DOE site, and the various storage configurations within the storage site.

Table 2–2 Sodium-Bonded Spent Nuclear Fuel Storage Locations and Configurations

<i>Spent Nuclear Fuel Type</i>	<i>Current Storage Locations and Configurations</i>		
	<i>DOE Site</i>	<i>Location</i>	<i>Configuration</i>
EBR-II driver	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Loose elements in canisters
		Hot Fuel Examination Facility	Loose elements
		Fuel Conditioning Facility	In process material ^a
EBR-II blanket	INEEL (ANL-W)	Radioactive Scrap and Waste Facility	Elements in canisters
		Fuel Conditioning Facility	In process material ^a
EBR-II driver	INEEL (INTEC)	CPP-603 basin	About 12 elements per canister
		CPP-666 basin	
Fermi-1 blanket	INEEL (INTEC)	CPP-749 dry well underground	Cut/uncut assemblies in 14 storage canisters
Fast Flux Test Facility driver	INEEL (ANL-W)	Hot Fuel Examination Facility	Loose elements
	Hanford	Fast Flux Test Facility, Buildings 405 and 403	Intact assemblies
Miscellaneous	Sandia National Laboratories/ New Mexico	Technical Area V	Experimental capsule
	SRS	Receiving Basin for Offsite Fuel	Elements
	Oak Ridge National Laboratory	Building 3525	Elements

^a Processed as part of the EBR-II Electrometallurgical Treatment Research and Demonstration Project.

2.3 TREATMENT AND MANAGEMENT METHODS

DOE has identified several potential treatment, management, and packaging methods that could be used to prepare sodium-bonded spent nuclear fuel for disposal in a geologic repository. These are: the electrometallurgical process; the PUREX process; packaging in high-integrity cans; the melt and dilute process; the glass material oxidation and dissolution system (GMODS) process; the direct plasma arc-vitreous ceramic process; and the chloride volatility process. Each of these methods is discussed below. In formulating reasonable alternatives under the proposed action, the GMODS process, the direct plasma arc process, and the chloride volatility process were not considered sufficiently mature technologies to be included as reasonable alternatives (see Section 2.6).

Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository without treatment, i.e., packaging the fuel in high-integrity cans without sodium removal, has been considered in this EIS under the No Action Alternative. The option may not meet current U.S. Nuclear Regulatory Commission (NRC) and/or Resource Conservation and Recovery Act (RCRA) requirements.

2.3.1 Electrometallurgical Treatment Process

The electrometallurgical treatment process was developed at the Argonne National Laboratory for processing EBR-II driver and blanket spent nuclear fuel assemblies with metallic fuel. The process has been demonstrated for the stainless steel-clad uranium alloy fuel used in that reactor. Modifications to the process are used for the treatment of oxide, nitride, and carbide sodium-bonded spent nuclear fuel. The electrometallurgical treatment process uses electrorefining, which is an industrial technology used to produce pure metals from impure metal feedstock (DOE 1996). Electrorefining has been used to purify metal for more than 100 years. **Figure 2-2** illustrates the various steps within the electrometallurgical treatment process at ANL-W.

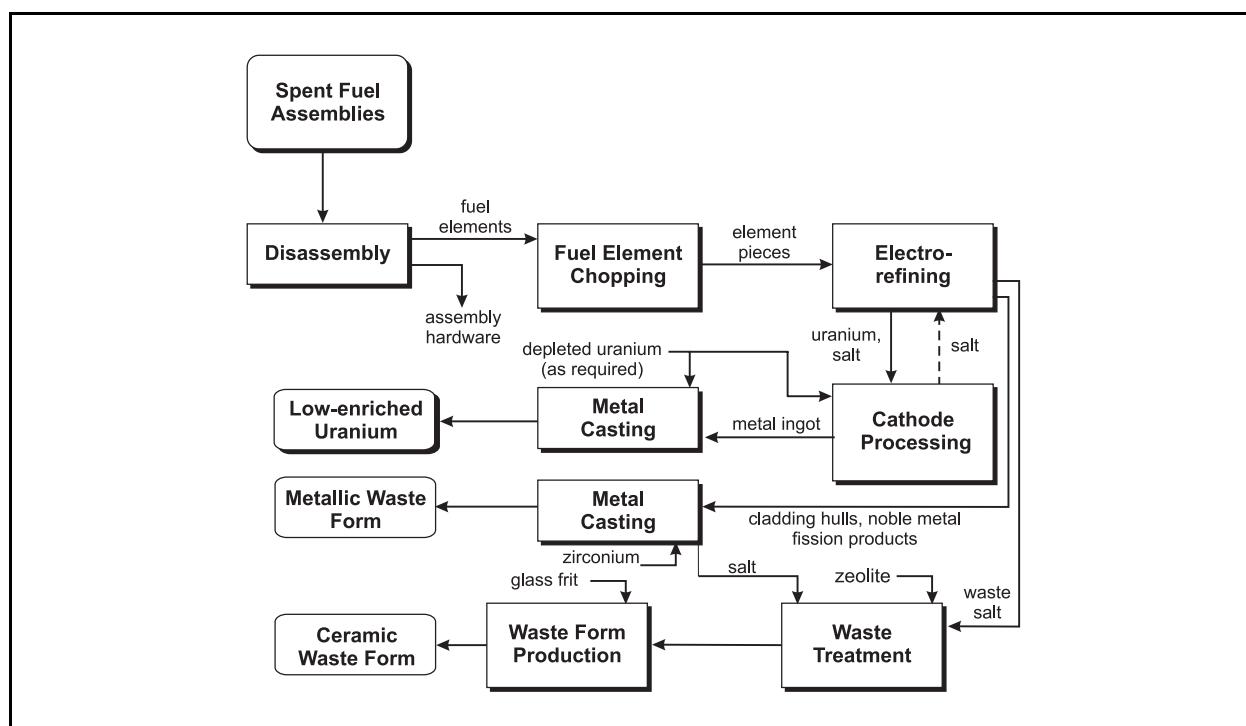


Figure 2-2 Electrometallurgical Treatment Process Flow Diagram

The first step in processing sodium-bonded spent nuclear metallic fuel involves the removal of fuel elements from the fuel assemblies. The fuel elements then would be chopped into short segments and placed in stainless steel baskets to form the anode in the electrorefiner.

The electrorefiner, in which the electrometallurgical treatment occurs, would be maintained at about 500 °C (930 °F) and contain a molten mixture of primarily two salts, lithium chloride and potassium chloride. The chopped fuel elements in the anode baskets would be lowered into the molten salt. Upon application of an electric voltage between the anodes and cathodes, uranium, transuranic elements including plutonium, most of the fission products, and the sodium would dissolve into the salt. The uranium would be deposited by the current at the cathode. The stainless steel cladding hulls and some of the insoluble fission products (i.e., noble metals) would remain in the anode baskets.

After a sufficient amount of spent nuclear fuel has been treated, the salt would be removed and solidified. The salt, which contains the sodium in the form of sodium chloride, transuranic elements, and most fission products extracted from the spent nuclear fuel, would be solidified, ground to a desired size, and mixed with zeolite. Zeolite is any of a group of alumina silicate minerals used as filters and ion-exchange agents. Zeolite is used to collect certain fission products from the process salt. The fission products, sodium, and transuranics, including plutonium in the salt and zeolite, would be heated so the salt becomes sorbed into the zeolite structure. Glass frit then would be added to the zeolite mixture and hot-pressed to produce a ceramic high-level radioactive waste form that is expected to be suitable for disposal.

The uranium deposited at the cathode would be removed from the electrorefiner and treated to remove any adhered salts. Then the uranium would be melted (and depleted uranium added if necessary), solidified to form an ingot, and further processed in a metal casting furnace to produce low-enriched uranium ingots. The stainless steel cladding hulls and the insoluble fission products would be melted in the casting furnace to produce a metal high-level radioactive waste form that is expected to be suitable for disposal in a geologic repository.

The oxide fuel would be prepared for treatment using the electrometallurgical treatment process by reducing the uranium oxide to uranium metal with lithium metal dissolved in small batches of lithium chloride-potassium chloride molten salt solution. The resulting uranium-bearing solution would be added to the molten salt solution used in the electrometallurgical treatment process for other sodium-bonded fuel and blanket elements and would be processed with those materials.

The carbide fuel would be prepared for electrometallurgical treatment by cleaning the fuel of sodium to the extent possible and then converting the fuel to uranium oxide with water or diluted acid. This oxide then would be converted to uranium metal by lithium metal in a molten salt solution and processed by electrometallurgical treatment with other sodium-bonded spent nuclear fuel and blanket elements. The nitride fuel also would be prepared for electrometallurgical treatment by converting it to uranium fuel.

In addition to the metal and ceramic waste form, some low-level radioactive waste also would be generated during the disassembly process of the spent nuclear fuel assemblies in the form of hardware. A detailed description of the electrometallurgical treatment process is presented in Appendix C.

2.3.2 PUREX Process

The PUREX process is a counter-current solvent extraction method which has been used extensively throughout the world since 1954 to separate and purify uranium and plutonium from fission products contained in aluminum-clad spent nuclear fuel and irradiated uranium targets. PUREX is not a thermal process; therefore, it takes place at low temperatures. DOE has two operating facilities at SRS, F-Canyon

and H-Canyon, that use the PUREX process for treatment of aluminum-clad fuel and targets. Use of the PUREX process facilities at SRS for treating sodium-bonded spent nuclear fuel involves certain restrictions inherent in the design: 1) the presence of sodium complicates the process as employed in the front-end of the SRS facilities; 2) the presence of stainless steel cladding would require significant modifications or additions to the existing front-end of the facilities; and (3) the presence of alloys (e.g., zirconium) in some of the fuel is incompatible with the SRS dissolution process. For this reason, treatment of driver sodium-bonded spent nuclear fuel is not feasible without significant modification to the existing PUREX process. However, the SRS facilities could be used without modification for the blanket sodium-bonded spent nuclear fuel if it is declad and its sodium removed prior to the process. In such a case, the F-Canyon facility would be used.

The fuel pins would be dissolved in an aqueous solution of nitric acid. The resulting nitric acid solution containing uranium, plutonium, and fission products would undergo feed clarification (to remove settleable solids) and acidity/alkalinity adjustment. The clarified aqueous solution then would be treated via the PUREX process utilizing centrifugal contactors and separators that involve organic solvent washing to produce: (1) an aqueous high-level radioactive waste containing the bulk of the fission products, americium, and neptunium; (2) a material stream containing the recovered plutonium; and (3) a material stream containing the recovered uranium. The plutonium- and uranium-containing streams each would undergo a second cycle of solvent washing to further separate the residual fission products and actinides from the plutonium and uranium. The aqueous high-level radioactive waste eventually would be processed to a borosilicate glass form. Material streams from the PUREX process would be uranium oxide, plutonium metal, and high-level radioactive waste. The uranium oxide would be stored on site as depleted uranium. The plutonium, approximately 260 kilograms (572 pounds), would be disposed of in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999c). **Figure 2–3** illustrates the various steps necessary for the treatment of sodium-bonded spent nuclear fuel in conjunction with the PUREX process. A detailed description of the process is presented in Appendix C.

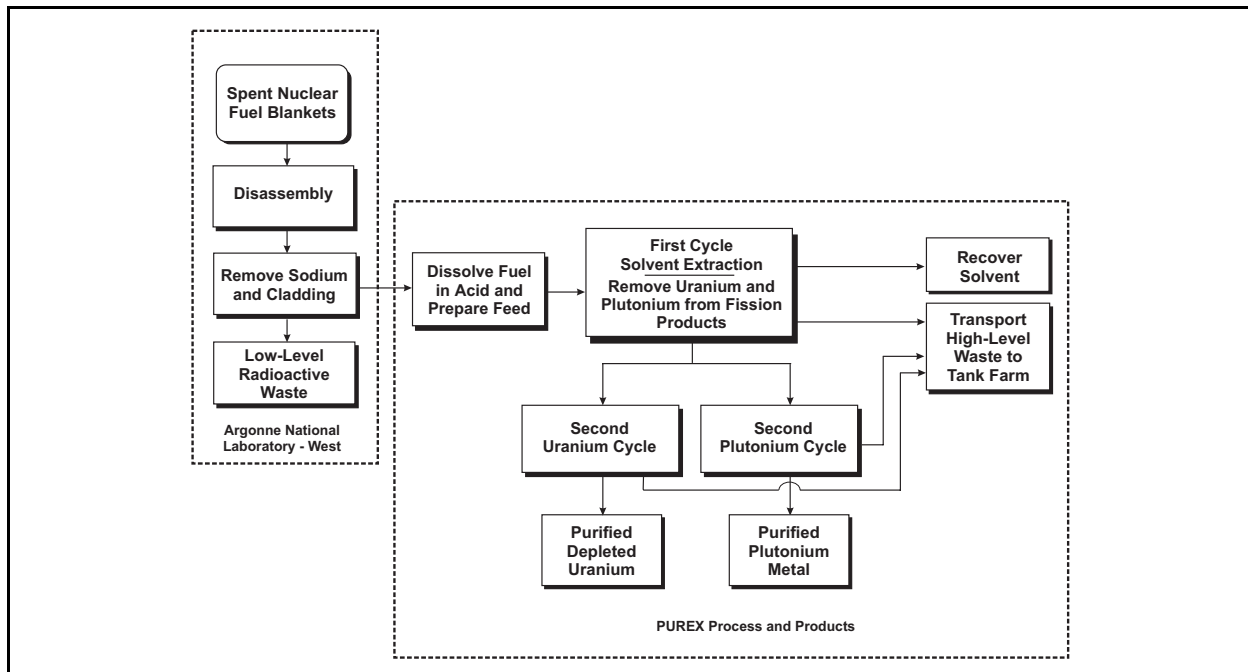


Figure 2–3 PUREX Process Flow Diagram

2.3.3 High-Integrity Can Packaging

The high-integrity can packaging provides substitute cladding for damaged or declad fuel, or another level of containment for intact fuel. The can would be used to store fuel on site until it is ready to be shipped to the repository. The can is constructed of a highly corrosion-resistant material (Hastelloy Alloy C-22) to provide corrosion protection during storage at the site. It also could provide long-term protection in a repository environment (i.e., for 1,000 or more years after repository closure with no institutional control). The high-integrity cans would be placed into a standardized canister for transportation and eventual placement in the repository in waste packages.

The analysis for packaging sodium-bonded spent nuclear fuel in high-integrity cans was performed with and without decladding and/or sodium removal. Packaging sodium-bonded blanket spent nuclear fuel in high-integrity cans with sodium removal was analyzed in the EIS under Alternative 2. Packaging sodium-bonded spent nuclear fuel in high-integrity cans without sodium removal was analyzed in this EIS as a direct disposal option under the No Action Alternative (see Sections 2.3.8 and 2.5.1).

The high-integrity cans would be placed in dry storage at ANL-W. Prior to emplacement in a repository, the high-integrity cans would be placed into a standardized canister designed to promote containment under repository conditions. **Figure 2-4** illustrates the high-integrity can flow process. A detailed description of the high-integrity can packaging is presented in Appendix C.

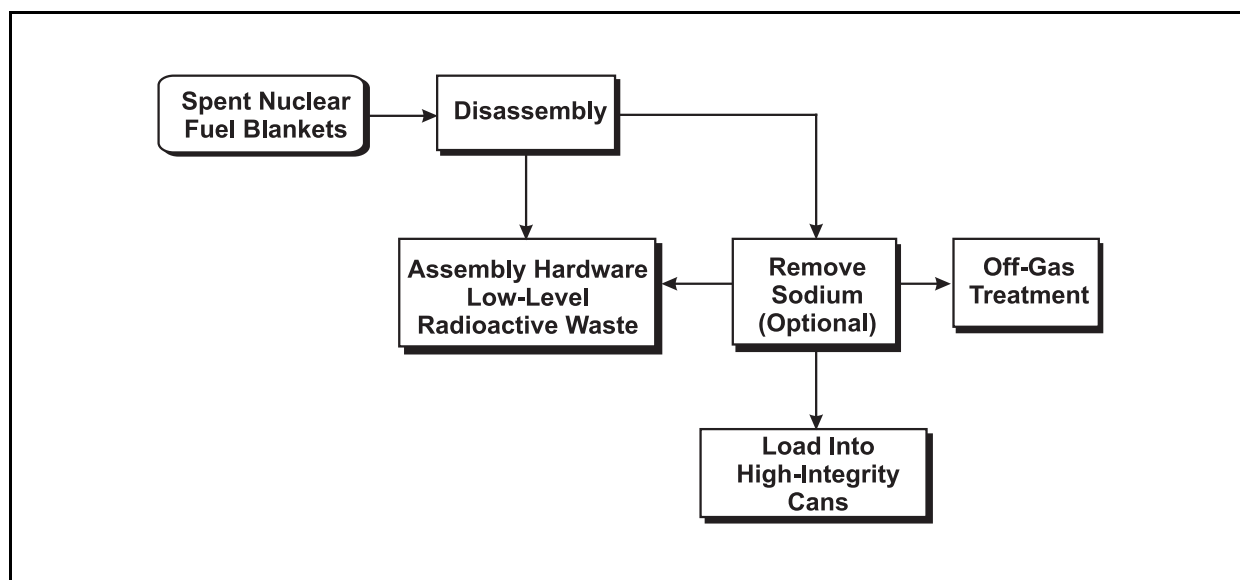


Figure 2-4 High-Integrity Can Packaging Flow Diagram

2.3.4 Melt and Dilute Process

There are three options for the melt and dilute process. In the first option, bare uranium blanket spent nuclear fuel pins with the sodium removed would be melted with aluminum at SRS using technology similar to that proposed for the aluminum-clad research reactor fuel. The second and third options would be conducted at the ANL-W site using metallurgical technology developed for uranium and stainless steel cladding. In the second option, blanket spent nuclear fuel elements would be melted with additional stainless steel. In the first two options, there would be no actual dilution of the fissile component of the uranium since it is present at only 0.2 percent, i.e., far less than the 0.7 percent in natural uranium. **Figure 2-5** illustrates the first two options of the melt and dilute process. The third option would involve a modified melt and dilute process

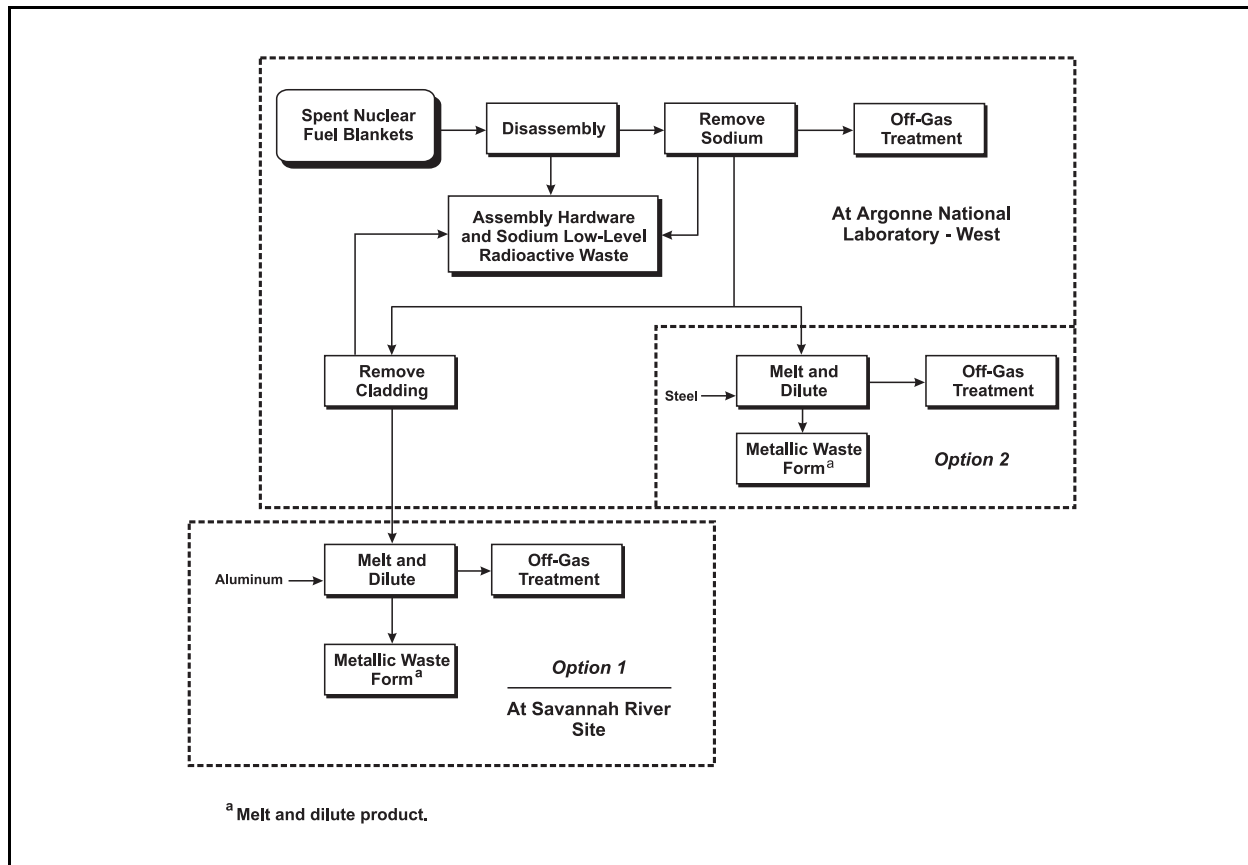


Figure 2-5 Melt and Dilute Process Flow Diagram (Options 1 and 2)

that would be capable of handling the sodium volatilized from processing chopped driver spent nuclear fuel elements with the cladding intact. **Figure 2-6** illustrates the third option of the melt and dilute process. A detailed description of the melt and dilute process is presented in Appendix C.

Option 1: Uranium-Aluminum Option for Blanket Pins

Declad and cleaned blanket pins would be received at SRS in aluminum canisters, each containing some 60 kilograms (132 pounds) of material. The canisters would be stored until they fit into the processing schedule. Following validation of the contents, the canisters would be loaded into a melting furnace with additional aluminum. The furnace would operate at a temperature of approximately 1,000 °C (1,830 °F), significantly in excess of the aluminum-uranium alloy melting temperature, to initiate melting within a reasonable time frame. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. A metal alloy ingot would be cast, sampled, and packaged.

Option 2: Uranium-Steel Option for Blanket Pins

Blanket elements with the sodium removed but not declad would be loaded into a furnace crucible. A small amount of radioactive waste steel might be added to the crucible to reach the desired uranium-steel composition. The furnace would be heated to approximately 1,400 °C (2,550 °F) to melt the uranium, after which the steel would be dissolved slowly into the uranium pool. The mixture would be stirred electromagnetically to a uniform composition. Volatile fission products would be captured by a series of filter banks before releasing the off-gas. A metal alloy ingot would be cast, sampled, and packaged.

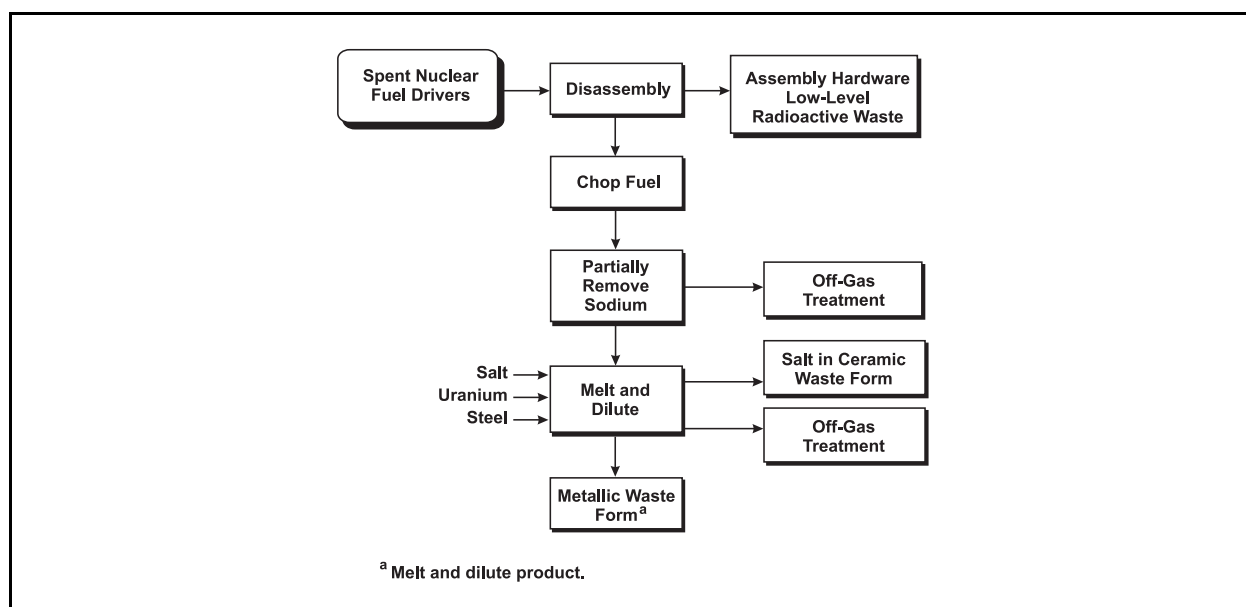


Figure 2-6 Melt and Dilute Process Flow Diagram (Option 3)

Option 3: Uranium-Steel Option for Driver Spent Nuclear Fuel

Some of the sodium in driver spent nuclear fuel elements would be removed in a similar manner to the sodium from blanket spent nuclear fuel elements. A modified melt and dilute process would be used for driver spent nuclear fuel still containing the cladding and some metallic sodium. The addition of flux salt is the only modification to the process required to capture residual sodium from the driver spent nuclear fuel. Chopped driver spent nuclear fuel elements would be loaded into an induction furnace and covered with a layer of low melting-temperature salt containing uranium, iron, or manganese chloride as a component to oxidize the molten sodium. Depleted uranium would be added in the ratio of about 2.5 to 1 to reduce the enrichment to less than 20 percent uranium-235. Radioactive waste steel would be added to complete the mixture. The use of radioactive waste steel reduces the inventory of the low-level radioactive waste. This furnace is operated at the same temperature as in Option 2. The sodium would react with and be captured in the flux salt, protecting the off-gas treatment filter banks. After the melt is mixed, a vacuum would be applied to complete the volatilization of the salt, which would be condensed and partially reused. The salt, which includes sodium in a nonreactive form, would be stabilized in a ceramic waste form similar to the waste form from the electrometallurgical treatment process. The metal melt would be stirred to achieve uniform composition and cast into an ingot, placed in a standardized canister, and stored.

The process described above can be used for sodium-bonded spent nuclear metallic fuel. Oxide, carbide, and uranium nitride fuel types cannot be treated using the melt and dilute process because of their high melting points.

2.3.5 GMODS Process

The GMODS process uses oxides to convert unprocessed spent nuclear fuel directly to borosilicate glass. The basic concept is to combine unprocessed sodium-bonded spent nuclear fuel and lead borate glass in a glass melter at a temperature of 800 to 1,000 °C (1,470 to 1,830 °F). The uranium and plutonium in the spent nuclear fuel would be converted into oxides and dissolved in the glass. Due to the powerful dissolution and oxidation properties of the lead borate glass melt, containment is a concern, and a water-cooled, cold-wall, induction-heated melter must be used. The waste form is borosilicate glass and would contain uranium,

the transuranic elements, the fission products, and the sodium present in the sodium-bonded spent nuclear fuel. As with all processes that dissolve or melt spent nuclear fuel, the GMODS process would produce radioactive off-gases. These gases would be filtered and treated by appropriate means. **Figure 2-7** illustrates the GMODS treatment process. A description of this process is presented in Appendix C.

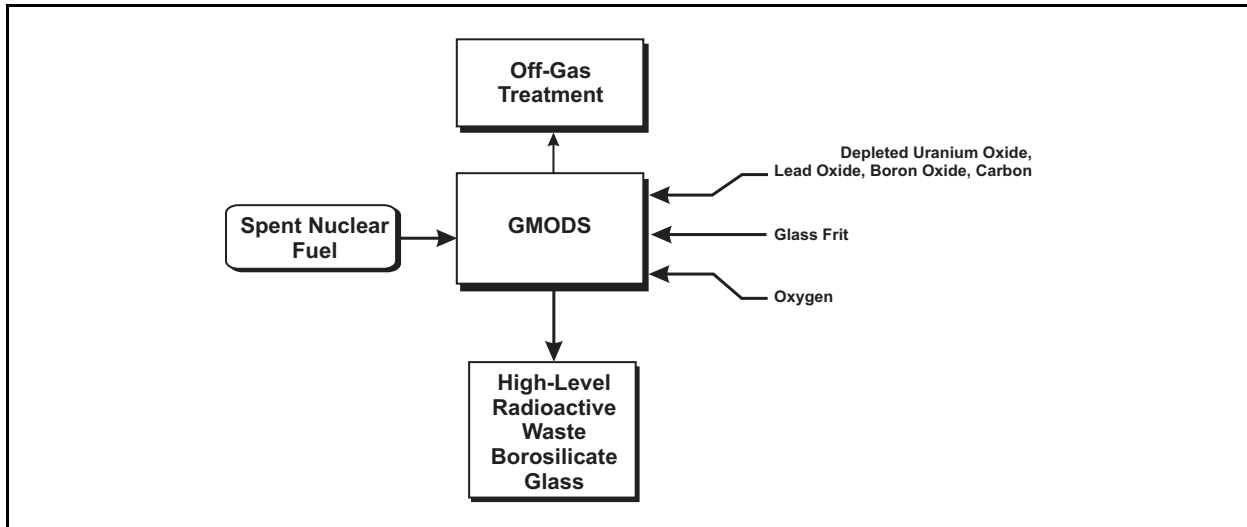


Figure 2-7 GMODS Process Flow Diagram

2.3.6 Direct Plasma Arc-Vitreous Ceramic Process

In this process, the sodium-bonded spent nuclear fuel would be cut into small pieces and melted and oxidized in a rotating furnace containing molten ceramic materials at a temperature of about 1,600 °C (2,900 °F) or higher. A direct-current plasma torch would supply the energy required in the process. Rotation would be used to keep the molten pool in the furnace. The spent nuclear fuel would be fed into the process with minimal pretreatment. Ceramic material would be added as necessary with the mixture homogenized by the torch. When the spent nuclear fuel is homogeneously melted and oxidized throughout the ceramic, rotation would be slowed to allow the molten vitreous ceramic to pour out by gravity flow into canister molds.

Metallic fuel such as the EBR-II spent nuclear fuel would require the addition of some ceramic material. Depleted uranium could be added to the process in almost any form to reduce the uranium-235 enrichment if necessary. Criticality issues would be addressed by limiting the process to batch runs of preselected quantities of fissile material with the addition of the depleted uranium and neutron poisons, if necessary.

As with all processes that dissolve or melt spent nuclear fuel, the plasma arc treatment would produce radioactive off-gases. These gases would be filtered and treated by appropriate means. **Figure 2-8** illustrates the direct plasma arc-vitreous ceramic treatment process. A description of this process is presented in Appendix C.

2.3.7 Chloride Volatility Process

The chloride volatility process is an advanced treatment technology that was investigated at INEEL. The process uses the differences in volatilities of chloride compounds to segregate major nonradiological constituents from spent nuclear fuel for the purpose of volume reduction, and isolates the fissile material to produce a glass or ceramic waste form. The major steps are: (1) high-temperature chlorination at about 1,500 °C (2,730 °F) and conversion of metallic fuel and cladding to gaseous chloride compounds; (2)

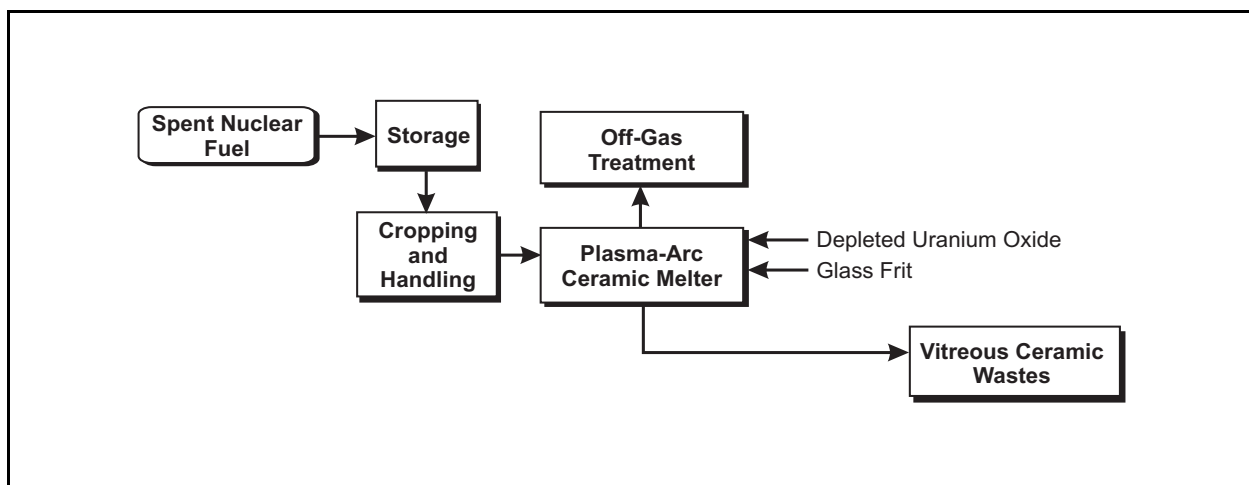


Figure 2-8 Direct Plasma Arc-Vitreous Ceramic Treatment Process Flow Diagram

removal of the transuranic chlorides and most of the fission products in a molten zinc chloride bed at approximately 400 °C (750 °F); (3) condensation of the other chlorides (e.g., uranium hexachloride) in a series of fluidized beds and condensers at successively lower temperatures; and (4) zinc chloride regeneration/recycling. The transuranics and fission product chlorides then would be converted into either fluorides or oxides for disposal. This process inherently handles volatilized fission products and chlorine gas, which presents significant unique occupational and public risks. **Figure 2-9** illustrates the chloride volatility treatment process. A description of this process is presented in Appendix C.

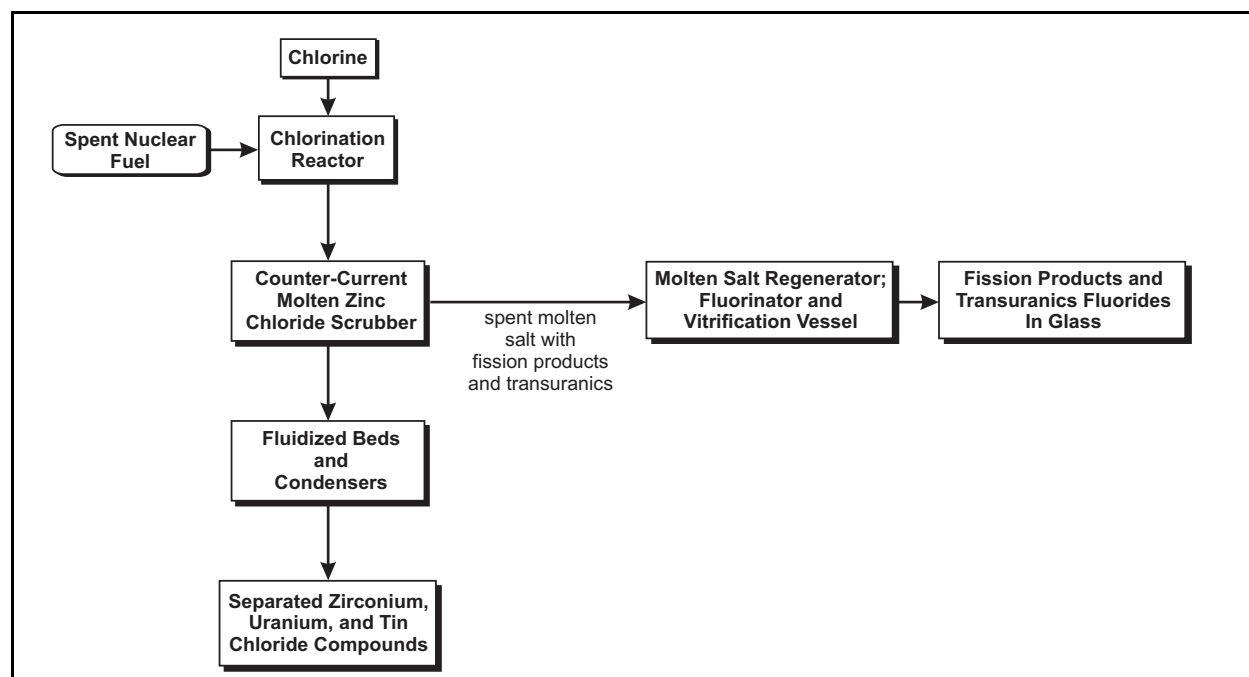


Figure 2-9 Chloride Volatility Process Flow Diagram

2.3.8 Direct Disposal

For the purpose of this EIS, direct disposal of sodium-bonded spent nuclear fuel is disposal without sodium removal. The sodium-bonded spent nuclear fuel (driver and blanket) would be packaged in high-integrity cans as described in Section 2.3.3 without removing the metallic sodium. The high-integrity cans would be placed into a standardized canister designed to provide containment under repository conditions during preclosure operations. At the present time, RCRA mixed waste (which contains both hazardous and radioactive waste) does not meet the requirements of acceptable waste as identified in the current April 1999 *Civilian Radioactive Waste Management System-Waste Acceptance System Requirements Document*, (DOE 1999a). Because of the presence of metallic sodium, the sodium-bonded spent nuclear fuel could be categorized as a RCRA hazardous waste that is potentially both pyrophoric and reactive (DOE 1997). Additionally, the NRC prohibits the disposal of materials that contain or generate explosive, pyrophoric, or chemically reactive substances that could compromise the repository's performance. Therefore, direct disposal would not meet current DOE or NRC repository acceptance criteria.

2.3.9 Sodium Removal and Disposition

As discussed in Section 2.2 and the preceding sections, the disposition of the metallic sodium in the sodium-bonded spent nuclear fuel varies with the treatment method. For those methods that do not require the removal of metallic sodium prior to treatment, or decladding of the fuel (e.g., the electrometallurgical process), the metallic sodium would be converted into a nonreactive salt as part of the process and would be incorporated in the high-level radioactive waste product of the process.

For the methods that require the removal of sodium prior to treatment and/or decladding of the spent fuel (i.e., the PUREX process, the melt and dilute process for blanket spent nuclear fuel [Options 1 and 2], and the packaging in high-integrity cans) the removed metallic sodium would be processed separately, converted into a nonreactive salt, and disposed of as low-level radioactive waste or high-level radioactive waste along with the waste form associated with the treatment process. Decladding and sodium removal could be done using either a mechanical process (the melt, drain, evaporate, and calcinate [MEDEC] process) or a laser declad and alcohol wash process.

In the MEDEC process, the blanket fuel is brought into an argon-atmosphere hot cell where the ends of the cladding for each fuel element are cut off to expose the sodium within the cladding. An argon-atmosphere is required for work involving materials such as sodium which could react with the moisture in air. Then the fuel is placed into a crucible furnace where it is subjected to a temperature of about 200 °C (390 °F), causing melting of the sodium, which is drained into a collection tank. After this bulk sodium is removed, the fuel temperature is raised to about 500 °C (930 °F) and a 10^{-4} torr vacuum is applied to the chamber housing this fuel. This higher temperature vacuum step volatilizes residual sodium and removes it from the fuel. This vacuumed sodium vapor would be condensed in a trap and collected with the previously removed bulk sodium pending further processing.

Sodium recovered during the cleaning process may contain some fission products, most notably cesium-137. The sodium would be stabilized using an oxidation/carbonation process (ANL 1999). Under this process, the cesium would be recovered by vacuum distillation of the sodium, taking advantage of the large difference in the boiling points of the two elements. The boiling point of cesium is 690 °C (1,274 °F), while the boiling point of sodium is 892 °C (1,638 °F). A vapor trap would be placed between the distillation column and pump to collect volatile species emitted from the condenser. The purified sodium would be processed by injection into a chamber, where it would react rapidly with oxygen and water to form aqueous sodium hydroxide. Carbon dioxide gas then would be bubbled through the hydroxide solution converting the sodium hydroxide to sodium carbonate. The aqueous sodium carbonate would be solidified with a binder and

packaged for disposal as low-level radioactive waste. The cesium fraction collected as distillate from the separation process would be added to the high-level radioactive waste form from the process.

To remove the cladding after sodium has been extracted, a special machine would be installed in the argon cell. This machine would mechanically push the fuel pins within the cladding out through the opening created when the cladding ends of the fuel elements previously were cut off. Experience with unirradiated blanket fuel at Argonne National Laboratory has shown that the pins could be mechanically pushed out of the stainless steel cladding after the sodium has been eliminated.

For the melt and dilute process for driver spent nuclear fuel (Option 3), the sodium removed prior to the process would be processed separately, converted into a nonreactive salt, and disposed of as low-level radioactive waste. Any sodium remaining within the fuel would be removed during the melt and dilute process as nonreactive salt, stabilized in a ceramic waste form, and disposed of as high-level radioactive waste.

The laser declad and alcohol wash process has been performed at Rockwell International Hot Laboratory in Canoga Park, California. The activities in this process would be similar to those that were performed at Rockwell International. The process would use a modified laser system for remote operation and a cutting machine to hold and index the fuel elements. The fuel elements would be brought into a hot cell and cut in a predefined cutting sequence. The fumes generated during the cutting process would be filtered and exhausted through an off-gas system. The fuel pins along with the cladding strips would be washed in an alcohol/water mixture to neutralize the metallic sodium and fission product (i.e., cesium) contamination. The fuel pins would be packaged and stored, or sent to SRS for treatment. The alcohol/water solution would be partially evaporated, and the sodium/cesium alcoholates and hydroxides would be neutralized, then solidified in a grouting agent, and disposed of as low- or high-level radioactive waste, depending on the cesium content.

Several aspects of the Rockwell laser process would not meet current environmental standards and would violate the design requirements of an argon hot cell. First, the Rockwell laser process required personnel entry into the hot cell on a biweekly basis for laser maintenance and purging of the cell atmosphere to maintain a low oxygen level (less than 4 percent) and to vent alcohol/water vapors and hydrogen gas from the cell. Neither of these practices would be acceptable for argon cell operation today, in part because of stricter radiation exposure controls and a higher concentration of fission products in the remaining inventory of EBR-II blanket fuel relative to the fuel that was treated by Rockwell. Operation of an argon cell requires maintenance of a low moisture and low oxygen content atmosphere as well as limitations on liquids within the cell for criticality control. The alcohol wash process introduces a liquid which is evaporated into the cell. Second, sodium collected during previous laser decladding operations was able to be disposed of as low-level radioactive waste. The sodium collected from processing the fuel addressed by this EIS would be contaminated with cesium. If sufficient quantities of cesium were present in the sodium, this waste could not be managed as low-level radioactive waste. For the sodium to be managed as low-level radioactive waste, the sodium would have to be processed (as is done with the sodium removed from the fuel in the MEDEC process) using a currently undefined process to remove the cesium from the alcohol mixture. While criticality concerns related to high moisture content levels within a multipurpose argon cell could be eliminated by removing any stored fissile materials, frequent purging of the hot cell atmosphere and personnel entry would still be restricted by current radiation exposure controls and the high concentration of fission products involved. Only the MEDEC process was used to evaluate the various alternatives that require cleaning and/or decladding of the sodium-bonded spent nuclear fuel because of compatibility concerns about laser operation in the cell.

Table 2–3 a summarizes sodium removal and disposition for the treatment methods addressed in this EIS.

Table 2–3 Sodium Removal and Disposition by Treatment and Management Method

<i>Treatment and Management Methods</i>	<i>Decladding</i>	<i>Sodium Treatment</i>	<i>Sodium Disposition</i>
Electrometallurgical process Driver and blanket fuel	No	Stabilization	Converted into nonreactive form, as part of the process, and disposed of with the high-level radioactive ceramic waste product of the process.
High-integrity cans Blanket fuel	No	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
PUREX process Blanket fuel	Yes	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Melt and dilute process Driver fuel	No	Removal	Part of the sodium is converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste. The remaining sodium is separated during the process, converted to nonreactive ceramic waste form, and disposed of as high-level radioactive waste.
Blanket fuel	Yes ^a /No ^b	Removal	Converted into nonreactive form, separately from the process, and disposed of as low-level radioactive waste.
Direct disposal ^c Driver and blanket fuel	No	No	Disposed of in metallic reactive form in high-integrity cans.

^a Melt and dilute process at SRS.^b Melt and dilute process at ANL-W.^c The direct disposal option may not meet current NRC and/or RCRA requirements.

2.4 SPENT NUCLEAR FUEL TREATMENT AND MANAGEMENT FACILITIES

For each alternative, DOE would use existing spent nuclear fuel management facilities that provide remote-handling and heavy-lifting capability, hot cells, and space to receive sodium-bonded spent nuclear fuel shipments. These facilities would prepare, treat, and/or place the sodium-bonded spent nuclear fuel in interim storage awaiting treatment as needed. Besides treating the sodium-bonded spent nuclear fuel, these facilities would provide capabilities to open the shipping containers, sample and analyze the fuel, and vacuum-dry the spent nuclear fuel. These facilities also could be used to repackage the fuel into storage canisters and place the repackaged fuel in dry interim storage to await treatment. The spent nuclear fuel management facilities described in the following sections provide the capability to implement the proposed action for each of the previously described technology alternatives.

2.4.1 ANL-W

The ANL-W site is a center of nuclear technology development and testing (**Figure 2–10**). Five nuclear test reactors have operated on the site, although the only one currently active is a small reactor used for radiography examination of experiments, waste containers, and spent nuclear fuel. Work on highly radioactive materials is conducted in the Fuel Conditioning Facility and the Hot Fuel Examination Facility, both heavily shielded hot cell facilities. Inventories of nuclear materials are maintained on site for conducting research, as well as for storage, pending decisions for further disposition.

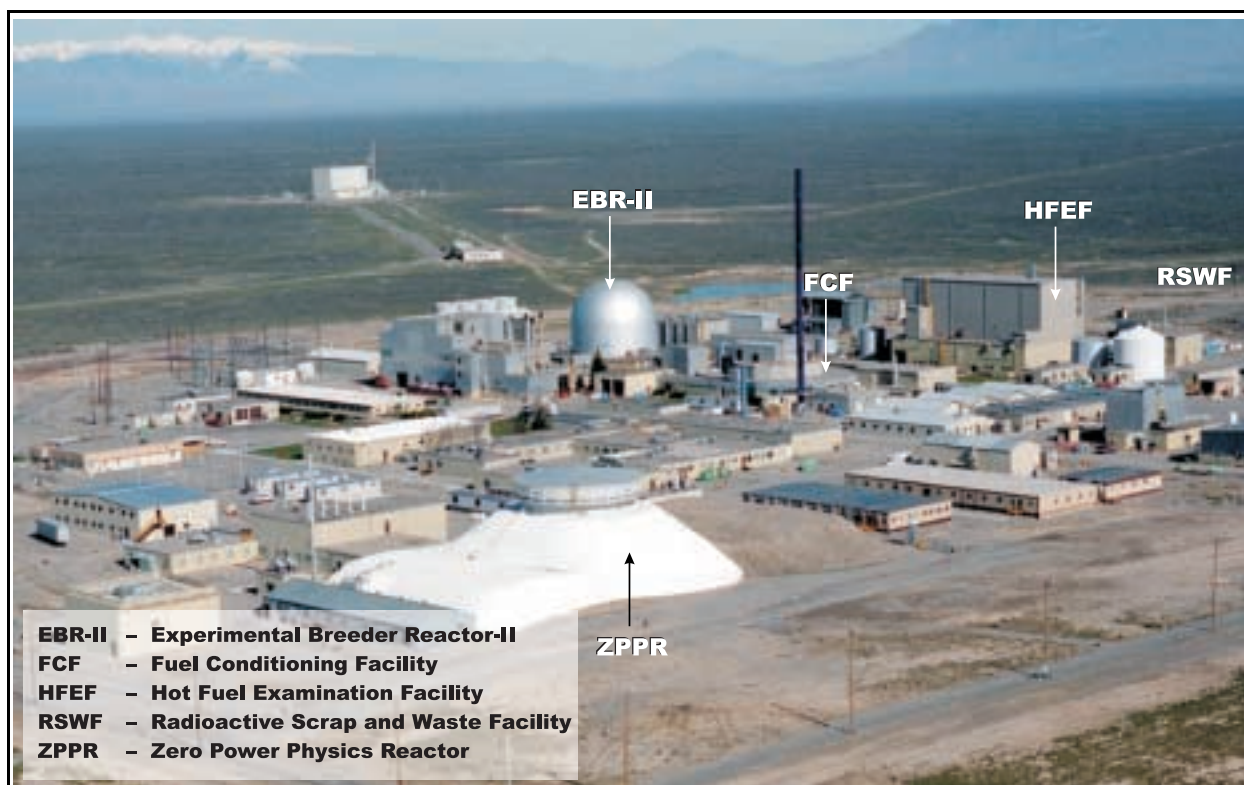


Figure 2–10 ANL-W

2.4.1.1 Fuel Conditioning Facility

The Fuel Conditioning Facility is one of the proposed facilities for the treatment and management of sodium-bonded spent nuclear fuel. The Fuel Conditioning Facility was activated in 1963 and consists of two hot cells, one with an air atmosphere and the other with an inert argon gas atmosphere. Since 1990, the Fuel Conditioning Facility has undergone major reconstruction and refurbishment to meet current safety and environmental requirements. A photographic view of the Fuel Conditioning Facility is shown in **Figure 2–11**. The hot cells enable technicians to work safely with radioactive nuclear materials from behind 1.5-meter-thick (5-foot-thick) shielding walls. A schematic plan of the main floor of the Fuel Conditioning Facility is shown in **Figure 2–12**. The rectangular air cell is used for handling, storage, and assembly/disassembly of components. The argon cell is a much larger hot cell and is “doughnut”-shaped; that is, personnel can work from the outside corridor around the hot cell and can monitor the work in the hot cell from an inner shielded work space in the center of the hot cell.

All equipment in the cells can either be repaired remotely using externally controlled robotic arms (manipulators) and cranes or can be removed and decontaminated for repair elsewhere in the facility from outside corridors around the hot cells.

In addition, the facility contains a mockup area where new equipment can be qualified and tested for remote operation and maintenance prior to installation in the cells. There is also a spray chamber, special glove boxes, and a suited-entry repair area (located in the basement) where contaminated equipment can be decontaminated and repaired.



Figure 2–11 Fuel Conditioning Facility at ANL-W

2.4.1.2 Hot Fuel Examination Facility

The Hot Fuel Examination Facility is one of the proposed facilities for the treatment and management of sodium-bonded spent nuclear fuel. The Hot Fuel Examination Facility is a hot cell complex built in the early 1970s for the preparation and examination of irradiation experiments in support of a wide variety of programs and process demonstrations. A photographic view of the Hot Fuel Examination Facility is shown in **Figure 2–13**. A wide range of remote operations and examinations may be performed in this facility with its shielded cells, support areas, and equipment.

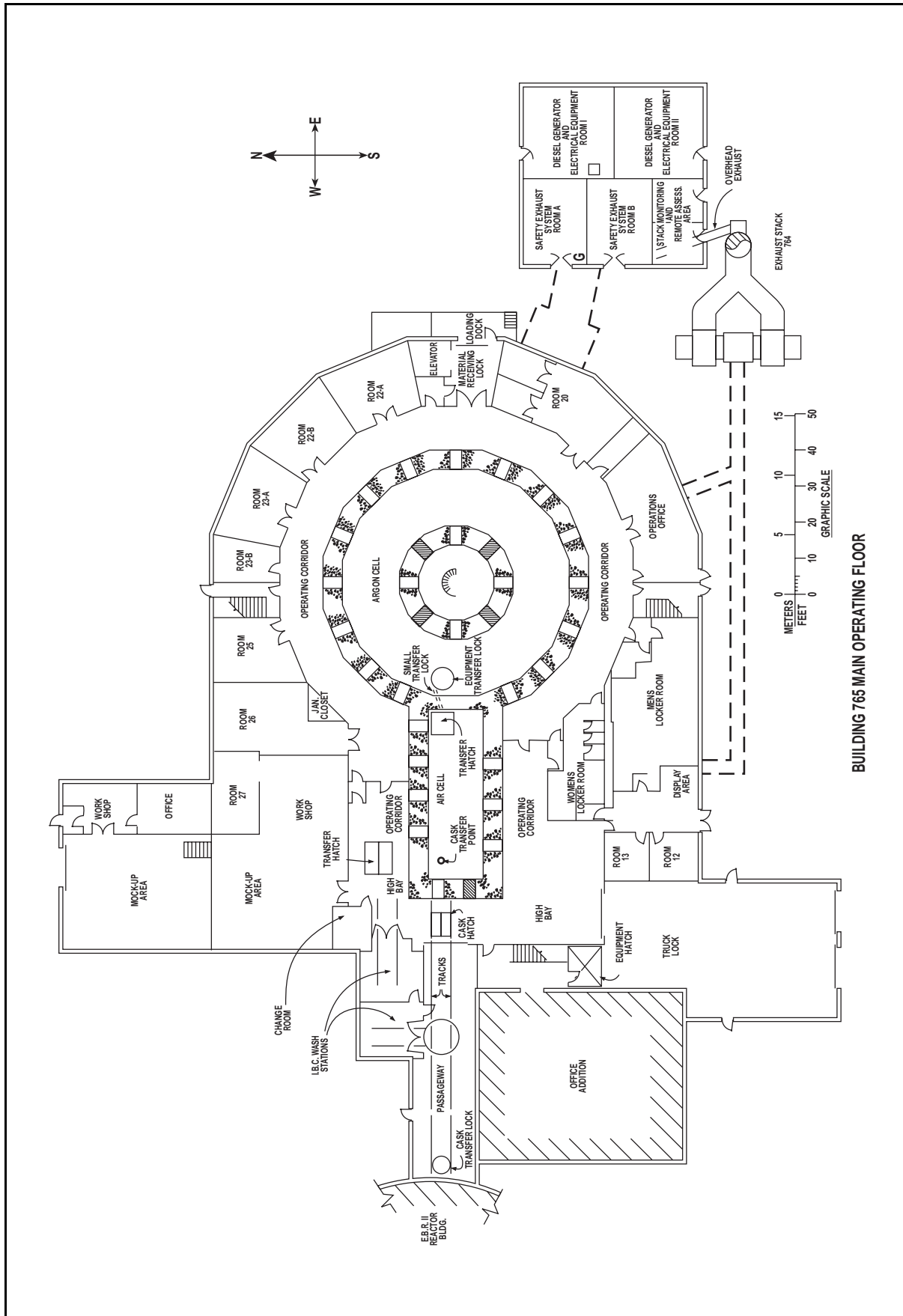


Figure 2-12 Main Floor Layout of the Fuel Conditioning Facility at ANL-W



Figure 2–13 Hot Fuel Examination Facility at ANL-W

The Hot Fuel Examination Facility is designed to be adapted to a wide variety of programs and consists primarily of two adjacent shielded cells, the main cell and the decontamination cell, in a three-story building. The decontamination cell contains an air atmosphere. A schematic plan of the main floor of the Hot Fuel Examination Facility is shown in **Figure 2–14**. The main cell contains an argon atmosphere for work involving materials such as sodium, plutonium, and other materials which could react with air. Both cells are surrounded by 1.2-meter-thick (4-foot-thick), high-density concrete to protect workers from the high radiation levels present in the hot cells. There are 21 work stations in the Hot Fuel Examination Facility, all equipped with shielded windows and remote manipulators. All in-cell equipment is carefully designed to permit remote operation and maintenance. A truck lock is located at the west end of the cell complex. The truck lock is large enough to accommodate various trucks and fork lifts which are used to transport shielded casks used in the day-to-day operation of the facility. The facility has recently been modified to accept truck-sized spent nuclear fuel shipping casks.

A high bay area covering the entire cell complex and serviced by a 40-ton bridge crane provides access from the truck lock to the top of the cells for bottom-opening casks. This area contains repair rooms, change rooms, and an access room and provides space for clean equipment repair and mockup.

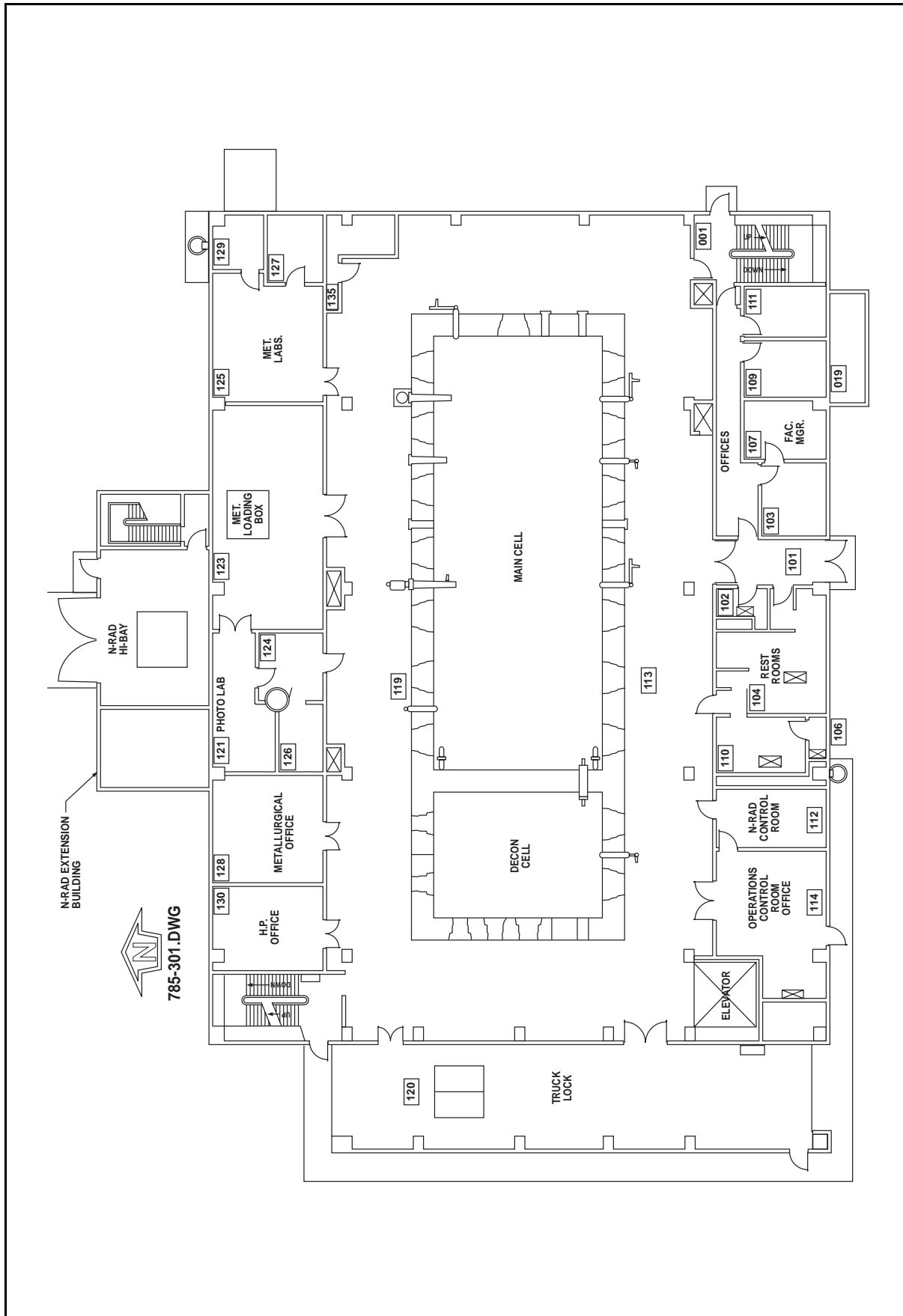


Figure 2-14 Main Floor Layout of the Hot Fuel Examination Facility at ANL-W

2.4.1.3 Zero Power Physics Reactor Materials Storage Building

The Zero Power Physics Reactor is currently shut down, but the facility is used for a number of projects, including a gas generation experiment. The Zero Power Physics Reactor Materials Storage Building, shown in **Figure 2–15**, is one of the primary storage facilities at ANL-W for uranium metal. Inventories of nuclear materials stored in this facility are maintained for conducting research as well as for storage, pending decisions for further disposition.



Figure 2–15 Zero Power Physics Reactor Materials Storage Building at ANL-W

2.4.1.4 Radioactive Scrap and Waste Facility

The Radioactive Scrap and Waste Facility, occupying about 1.6 hectares (4 acres), provides safe interim dry storage for spent nuclear fuel and waste generated from experiments. It is one of the facilities where sodium-bonded spent nuclear fuel currently is stored and the facility where high-level radioactive waste from the treatment of the fuel could be stored pending ultimate disposal. It is located underground and 0.8 kilometers (0.5 miles) northeast of the main ANL-W facilities within the ANL-W boundary. The Radioactive Scrap and Waste Facility looks somewhat like a large parking lot on the surface, as shown in **Figure 2–16**. The facility has a permit issued by the State of Idaho for interim storage of mixed waste regulated under RCRA. The Radioactive Scrap and Waste Facility provides protection against corrosion for the more than 1,000 underground steel liners available for waste storage of materials handled at ANL-W.



Figure 2–16 Radioactive Scrap and Waste Facility at ANL-W

2.4.2 INTEC at INEEL

INTEC is located 20 kilometers (12.4 miles) west-southwest of ANL-W. It is one of the sites where sodium-bonded spent nuclear fuel currently is stored. A photographic view of INTEC is shown in **Figure 2–17**. INTEC was constructed in the 1950s to reprocess spent nuclear fuel from government reactors. In 1992, DOE announced that it no longer would reprocess spent nuclear fuel. Current work at INTEC includes receiving and storing spent nuclear fuel, converting liquid sodium-bearing waste to granular solid, environmental restoration and decontamination and dismantling activities, and technology development. About 880 people currently work at INTEC. The facility would be used to continue storing sodium-bonded spent nuclear fuel and for packaging the treated sodium-bonded spent nuclear fuel in standardized canisters in preparation for transport and disposal in a geologic repository. However, because it has no hot cell with an inert gas atmosphere, it cannot be used for any sodium removal activities under the proposed action.



Figure 2-17 INTEC at INEEL

The primary facilities at INTEC include:

- The Fluorinel Dissolution Process and Fuel Storage Facility. This facility is divided into two parts, a spent nuclear fuel storage area and the Fluorinel Dissolution Facility. The storage area consists of six storage pools for storing nuclear fuel. Radioactive spent nuclear fuel is stored under about 11 million liters (3 million gallons) of water, which provide protective shielding and cooling. Eventually, all spent nuclear fuel will be removed from underwater storage pools and placed in a dry storage system and prepared for shipment to a repository.

The spent nuclear fuel, from government-owned reactors, was formerly reprocessed at INTEC to recover reusable uranium. The Fluorinel Dissolution Facility includes an air atmosphere “hot cell” with 1.8-meter-thick (6-foot-thick) concrete walls where spent nuclear fuel was dissolved in an acid solution. With the end of reprocessing, uranium and hazardous materials were flushed from the Fluorinel Dissolution Facility. New missions for this facility are under consideration.

- The 603 Fuel Storage Building. This building houses both underwater pools and dry storage facilities for spent nuclear fuel. The pools were constructed in the 1950s and served as the primary spent nuclear fuel storage facility until the Fluorinel Dissolution Process and Fuel Storage Facility opened in 1984. Fuel in underwater storage at Building 603 is being transferred to the newer storage pools at the Fluorinel Dissolution Process and Fuel Storage Facility. Also located in the building is the Irradiated Fuel Storage Facility, which stores dry fuel that cannot be stored underwater. The Irradiated Fuel Storage Facility has 636 storage positions, with 297 in use. The majority of the spent nuclear fuel stored in this facility came from the Fort St. Vrain commercial reactor in Colorado.
- The New Waste Calcining Facility. This facility converted liquid high-level radioactive waste into a granular solid similar in consistency to dry laundry detergent. The liquid waste was drawn from underground storage tanks and sprayed into a vessel superheated by a mixture of kerosene and oxygen. Most of the liquid evaporated, while radioactive fission products adhered to the granular bed material in the vessel. The off-gases were treated and monitored before they were released to the environment, and the residual solids were transferred to large stainless steel structures encased in thick concrete vaults. This conversion process achieved an 8-to-1 volume reduction from liquid to solid. The same process currently is used to convert sodium-bearing waste to granular solid with a smaller volume reduction from liquid to solid.
- The Remote Analytical Laboratory. This laboratory is designed for the safe examination of radioactive samples to support the New Waste Calcining Facility mission and other INTEC operations. The facility includes a 356-square-meter (3,500-square-foot) air atmosphere hot cell with 90-centimeter (3-foot) concrete shield walls that allow remote examination of radioactive chemical samples.
- The INTEC 601/602 Processing Corridors. The processing corridors were used to extract highly enriched uranium from dissolved spent nuclear fuel during reprocessing and to solidify the recovered highly enriched uranium for shipment off site. Built in the 1950s, these facilities were to be replaced by the Fuel Processing Restoration Project. Because DOE decided to discontinue reprocessing, these facilities have been flushed to remove uranium and hazardous materials. The Fuel Processing Restoration Project, about 40 percent complete when construction stopped in 1992, was discontinued in a manner that preserves the facility for possible use in future research and development missions at INTEC.
- The High-Level Radioactive Waste Tank Farm. The waste tank farm includes 11 underground stainless steel storage tanks used to store the radioactive liquid waste generated during the reprocessing of spent

nuclear fuel and plant decontamination work. DOE is evaluating options for treating the remaining Tank Farm liquid waste in the *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement*, which was issued for public comment in December 1999 (DOE 1999b). The underground tanks are encased in concrete vaults which have sumps and leak detectors. One tank is always kept empty for use as a transfer backup should a problem develop with one of the other 10 tanks. The tanks are corrosion-resistant, and no leakage has been detected. Some leaks from transfer lines outside the tanks have occurred, however, and the contaminated soil is scheduled for environmental cleanup.

2.4.3 SRS

SRS was constructed during the early 1950s to produce the basic materials used in the fabrication of nuclear weapons, primarily tritium and plutonium-239. Five reactors were built on the site. The reactors produced nuclear materials by irradiating target materials with neutrons. Also built were support facilities including two chemical separation plants, a heavy water extraction plant, a nuclear fuel and target fabrication facility, and waste management facilities.

Irradiated materials were moved from the reactors to the two chemical separation facilities—the next step in the production process. In these facilities, known as “canyons,” the irradiated fuel and target assemblies were processed chemically to separate useful products from waste. After refinement, some nuclear materials were shipped to other DOE sites for final use.

SRS has adjusted through the years to meet changing defense requirements. All five of the original SRS production reactors are permanently shut down. While production of new tritium will not be necessary for several years, recycling and reloading of tritium to maintain nuclear weapons’ reliability is a continuing site mission.

DOE currently uses the F-Canyon chemical separation facility and the FB-Line to stabilize spent nuclear fuel, as described in the *Final Environmental Impact Statement, Interim Management of Nuclear Materials* (DOE 1995b). The *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000) will be used to help determine the most appropriate final disposition option for spent nuclear fuel currently assigned to SRS.

Weapons material production at SRS has produced unusable by-products such as high-level radioactive waste. The high-level radioactive waste, approximately 35 million gallons, is stored in waste tanks on site. The Defense Waste Processing Facility will bond the radioactive elements in borosilicate glass, a stable form for disposal.

2.4.3.1 F-Canyon

The F-Canyon at SRS could chemically separate uranium from fission products in blanket spent nuclear fuel using the PUREX process. A photographic view of the F-Canyon complex is shown in **Figure 2–18**. The canyon facilities use radiochemical processes for the separation and recovery of plutonium and uranium isotopes. Historically, F-Canyon recovered plutonium-239 and uranium-238 from irradiated natural or depleted uranium.

The Canyon building is a reinforced concrete structure, 254 meters (835 feet) long by 37 meters (122 feet) wide by 20 meters (66 feet) high. It houses large equipment (tanks, process vessels, evaporators, etc.) used in the chemical separation processes.



Figure 2–18 The F-Canyon Complex at SRS

The F-Canyon facility actually contains two canyons, the hot canyon and the warm canyon, as shown in **Figure 2–19**. The two canyons are parallel and separated by a center section, which has four floors. The center section contains office space, the control room for facility operations, chemical feed systems, and support equipment such as ventilation fans. Processing operations involving high radiation levels (dissolution, fission product separation, and high-level radioactive waste evaporation) occur in the hot canyon, which has thick concrete walls to shield people outside and in the center section from radiation. The final steps of the chemical separation process, which generally involve lower radiation levels, occur in the warm canyon.

2.4.3.1.1 FB-Line

The FB-Line, located on top of the F-Canyon, historically converted plutonium nitrate solution produced in the F-Canyon to plutonium-239 metal buttons. Solutions from the F-Canyon are concentrated and purified in the FB-Line. The plutonium then is precipitated, filtered, dried, and finally reduced to a metallic form called a button. The button is about the size of a hockey puck. Processing equipment is enclosed in glove boxes so that employees and operating areas are not exposed to the radioactive material. Some of these operations are automated. The FB-Line also recycles plutonium scrap generated from facility operations and offsite sources. In September 1997, the FB-Line began a new plutonium packaging process. This process places stabilized plutonium in rugged, welded stainless steel cans. DOE also has determined that the FB-Line should be used to stabilize plutonium.

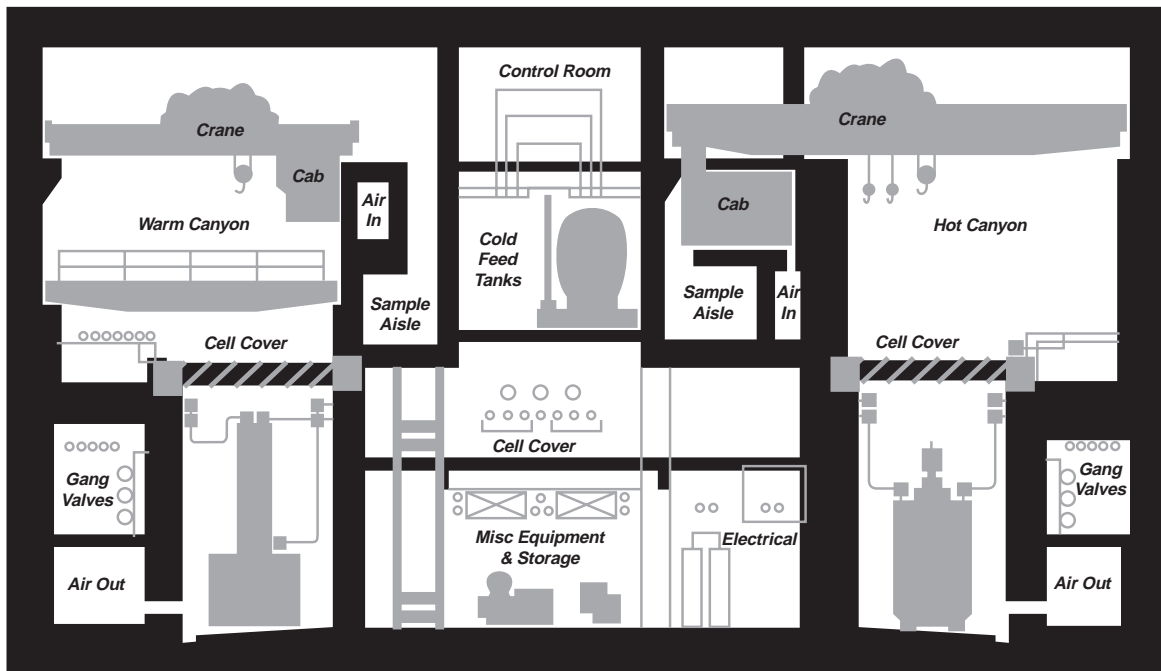


Figure 2-19 F-Canyon Building Sections (Hot Canyon and Warm Canyon) at SRS

2.4.3.2 Building 105-L

Building 105-L is the SRS facility where installation of a melt and dilute process for treating spent nuclear fuel is proposed. Building 105-L is part of the currently shut-down L-Reactor complex at SRS. The L-Reactor was built in the early 1950s to produce nuclear materials for national defense. In 1988, DOE shut the reactor down for safety upgrades. At the completion of the upgrades, the reactor was not restarted. A photographic view of Building 105-L is shown in **Figure 2-20**. In 1993, DOE ended the reactor's materials production mission. The current mission of this facility is to store reactor components and other radioactive materials in the disassembly basin; receive and store foreign and domestic research reactor fuel in the disassembly basin; decontaminate shipping casks in the Building 105-L stack area; store contaminated moderators in tanks or drums; and compact low-level radioactive waste in a compactor. DOE maintains the structures, systems, and components necessary to perform these missions, but has de-energized, drained, or otherwise deactivated many others.

Building 105-L has space potentially suitable for installation of a melt and dilute process (DOE 2000). The space includes the process room and crane maintenance area. The process room, a shielded area situated above the reactor tank, formerly provided access to the reactor by means of a charge and discharge machine for handling reactor fuel assemblies. An overhead crane services the area. The crane maintenance area, connected to the process room by a shielded crane wash area, allows hands-on maintenance of the fuel assembly transfer systems. The Building 105-L stack area would be used to unload shipping casks from their containers and to decontaminate empty shipping casks.

In the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement*, (DOE 2000), DOE identified melt and dilute as one of the preferred methods of treating spent nuclear fuel at SRS. To implement the melt and dilute technology, DOE would construct a melt and dilute facility in the existing Building 105-L at SRS and build a dry storage facility in L-Area, near Building 105-L.



Figure 2–20 L-Reactor Complex at SRS

DOE expects the melt and dilute option would be relatively simple to implement in Building 105-L. The major technical issue in implementing this technology would be the design of an off-gas system to capture volatilized fission products. Preliminary engineering studies indicate that the system could be designed using proven approaches for managing off-gases. The impacts from the construction of a melt and dilute facility at SRS’s Building 105-L are addressed in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).

2.4.3.3 Defense Waste Processing Facility

The Defense Waste Processing Facility, located in the S-Area, converts high-level radioactive liquid waste currently stored at SRS into a solid glass form suitable for long-term storage and disposal. A photographic view of the Defense Waste Processing Facility is shown in **Figure 2–21**. This process, called “vitrification,” immobilizes high-level radioactive liquids into a more stable form suitable for disposal in a geologic repository. About 125 million liters (34 million gallons) of high-level liquid radioactive waste currently is stored in 49 underground carbon steel tanks at SRS. This waste has about 480 million curies of radioactivity, and requires permanent isolation from the environment.



Figure 2-21 Defense Waste Processing Facility at SRS

Construction of the Defense Waste Processing Facility began in 1983. Changing environmental requirements; major safety upgrades and process modifications; and a “waste qualification” test to demonstrate that the glass form meets all environmental and operational requirements for long-term storage were required before system testing began in 1990. The Defense Waste Processing Facility successfully completed its waste qualification testing in late 1995 and began operating in March 1996.

The Defense Waste Processing Facility treats the highly radioactive material removed from the original waste. In this process, a sand-like borosilicate glass is mixed with the waste and sent to the plant's steel and ceramic melter. In the melter, electricity is used to heat the waste/borosilicate glass mixture until molten. This molten glass-waste mixture is poured in a pencil-thin stream into stainless steel canisters to cool and harden. Each canister is approximately 3 meters (10 feet) tall and 0.6 meters (2 feet) in diameter; it takes approximately 24 hours to fill one canister. A filled Defense Waste Processing Facility canister weighs about 2,270 kilograms (5,000 pounds). The exterior of each canister is blasted with borosilicate glass to remove contamination, then welded shut after a plug has been rammed into place.

A specially designed “Shielded Canister Transporter” moves each sealed canister, one at a time, from the Defense Waste Processing Facility to the temporary storage building adjacent to the facility. This transporter is a two-wheel drive vehicle powered by redundant diesel engines. It has a center module with a shielding cask, floor plug cavity, and associated cask lifting equipment.

At the storage building, canisters are lowered by the transporter into an underground reinforced concrete vault containing 2,286 individual canister supports. This seismically qualified storage vault can hold, at current Defense Waste Processing Facility production flow, canisters from about 8 to 10 years of processing. More storage

buildings can be built according to the need for storage space. The canisters are stored at SRS until a Federal repository can be established.

2.5 ALTERNATIVES EVALUATED

As discussed in Section 2.1, the proposed action is to treat and manage sodium-bonded spent nuclear fuel. The alternatives under the proposed action are illustrated in **Figure 2–22** and are addressed below. Although each alternative addresses both driver and blanket spent nuclear fuel, DOE will consider the driver and blanket spent nuclear fuel, as well as the EBR-II and Fermi-1 blanket spent nuclear fuel, separately in identifying a preferred alternative and any subsequent Record of Decision. In other words, DOE is considering all combinations of technologies, options, and fuel types, not only the specific combinations that are explicitly discussed in the EIS. For example, “no action” may be chosen for the driver spent nuclear fuel and “melt and dilute at SRS” for the blanket spent nuclear fuel.

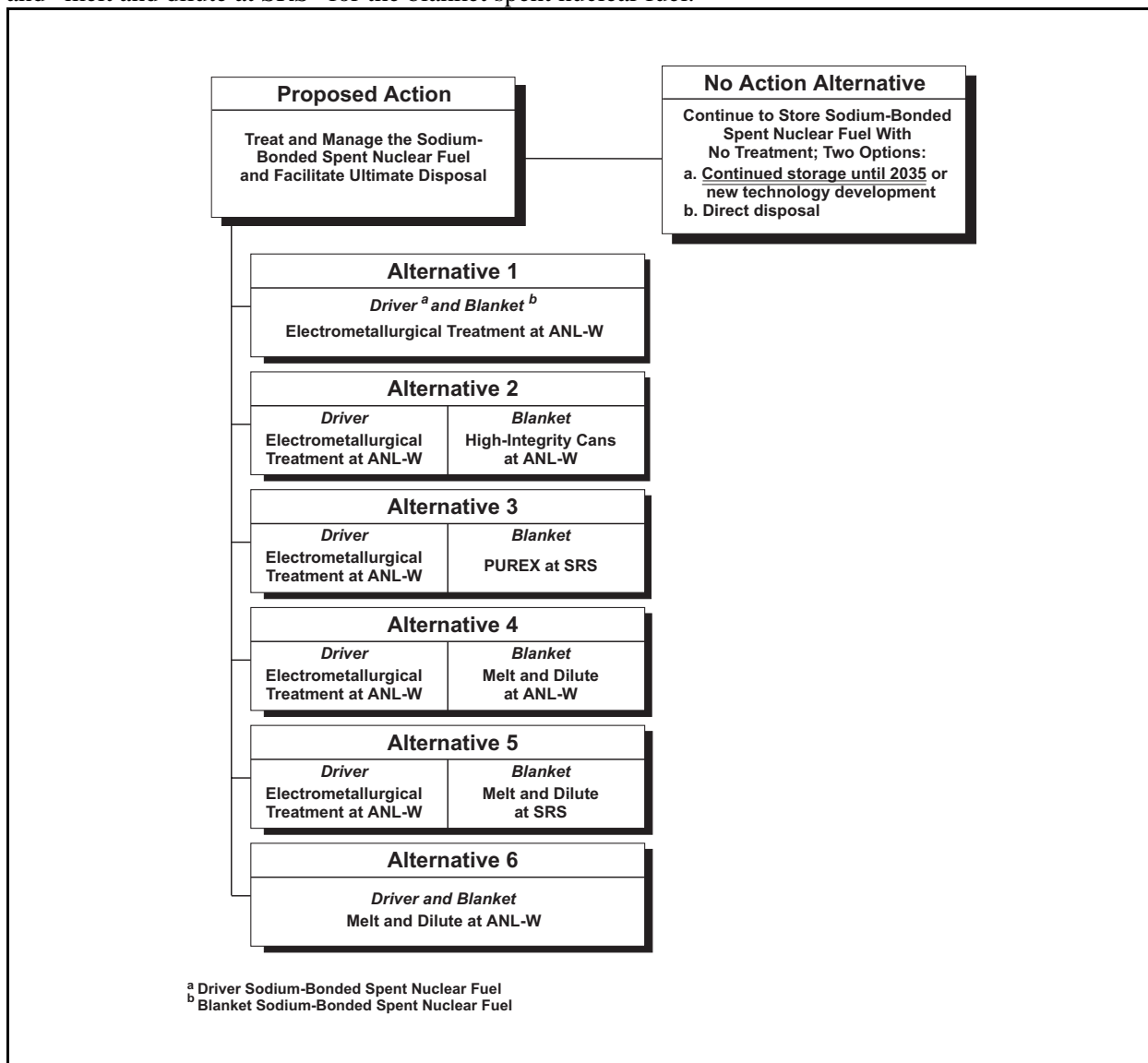


Figure 2–22 Proposed Action and Alternatives

2.5.1 No Action Alternative

Under the No Action Alternative, all or part of the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed), except for stabilization activities that may be necessary to prevent potential degradation of some of the spent nuclear fuel. Under the No Action Alternative, two options were analyzed: (1) the sodium-bonded spent nuclear fuel would continue to be stored until 2035 at its current location subject only to activities dictated by the amended Record of Decision (61 FR 9441) for the Programmatic Spent Nuclear Fuel EIS (DOE 1995a) and other existing site-specific NEPA documentation or until a technology currently dismissed as an unreasonable alternative because it is less mature (e.g., GMODS or plasma arc) is developed; and (2) the sodium-bonded spent nuclear fuel would be disposed of directly in a geologic repository without treatment. The fuel would be packaged in high-integrity cans without sodium removal. Under the latter option, the sodium-bonded spent nuclear fuel at INTEC would be transported to ANL-W for packaging. Both options would require the installation of some new waste handling equipment. As discussed in Section 2.3.8, the direct disposal option would not meet current DOE or NRC repository acceptance criteria requirements.

A fundamental assumption made under the No Action Alternative is that the sodium-bonded spent nuclear fuel eventually will be disposed of in a manner similar to the rest of the spent nuclear fuel owned by DOE and within the time period considered over which institutional controls could reliably be assumed to be in effect. In the event that the sodium-bonded spent nuclear fuel has not been treated before 2035, the temporarily stored fuel will be removed from the State of Idaho by the year 2035. The environmental impact of the removal of untreated sodium-bonded spent nuclear fuel would be evaluated in a separate NEPA document. The continued storage of untreated sodium-bonded spent nuclear fuel in the State of Idaho or elsewhere, beyond time periods for which institutional controls could reliably be assumed to be in effect, could lead to significant impacts to the environment and the health and safety of the public from radioactive releases caused by the gradual degradation of the fuel and its containment.

In selecting the No Action Alternative, DOE could actively pursue research and development of another treatment technology including, for example, the GMODS and plasma arc methods. These methods offer the potential for treating both driver and blanket spent nuclear fuel. They do not involve separation of uranium or plutonium, and the treatment product is expected to be suitable for disposal in a geologic repository. Reasons for not including these methods among the reasonable alternatives under the proposed action are provided in Section 2.6.

2.5.2 Alternative 1: Electrometallurgically Treat Blanket and Driver Fuel at ANL-W

Under this alternative, all sodium-bonded spent nuclear fuel (both driver and blanket, approximately 60 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process. **Figure 2–23** illustrates the steps of the process under Alternative 1.

The sodium-bonded spent nuclear fuel (driver and blanket) from ANL-W (the Radioactive Scrap and Waste Facility and the Hot Fuel Examination Facility) would be transported directly to the Fuel Conditioning Facility for electrometallurgical treatment. Spent nuclear fuel currently stored at INTEC would be transported to the Hot Fuel Examination Facility. This is necessary because only the Hot Fuel Examination Facility at ANL-W is capable of accepting spent nuclear fuel transportation casks. At the Hot Fuel Examination Facility, the fuel would be separated from the assembly hardware and packaged and transferred to the Fuel Conditioning Facility for electrometallurgical treatment. The separated hardware would be packaged and managed as low-level radioactive waste.

After treatment, the low-enriched uranium by-product from the cathode processing would be metal-casted at the Fuel Conditioning Facility and transferred to the Zero Power Physics Reactor Materials Storage Building for storage. The cladding hulls remaining at the anode would be packaged and transferred to the

Hot Fuel Examination Facility for metal casting into high-level radioactive waste and transferred to the Radioactive Scrap and Waste Facility for storage. The electrolyte salt containing the fission products, sodium, and transuranic elements would be transferred in metal cans back to the Hot Fuel Examination Facility where the ceramic waste would be produced. The ceramic waste cylinders would be packaged and transferred to the Radioactive Scrap and Waste Facility for storage. Implementing this alternative at the Fuel Conditioning Facility and the Hot Fuel Examination Facility would require the installation of some new waste handling equipment at the facilities. Electrometallurgical treatment of the sodium-bonded spent nuclear fuel at ANL-W could start as early as the year 2000, and would require approximately 13 years to process all fuel. Driver spent nuclear fuel alone would require approximately 7 years.

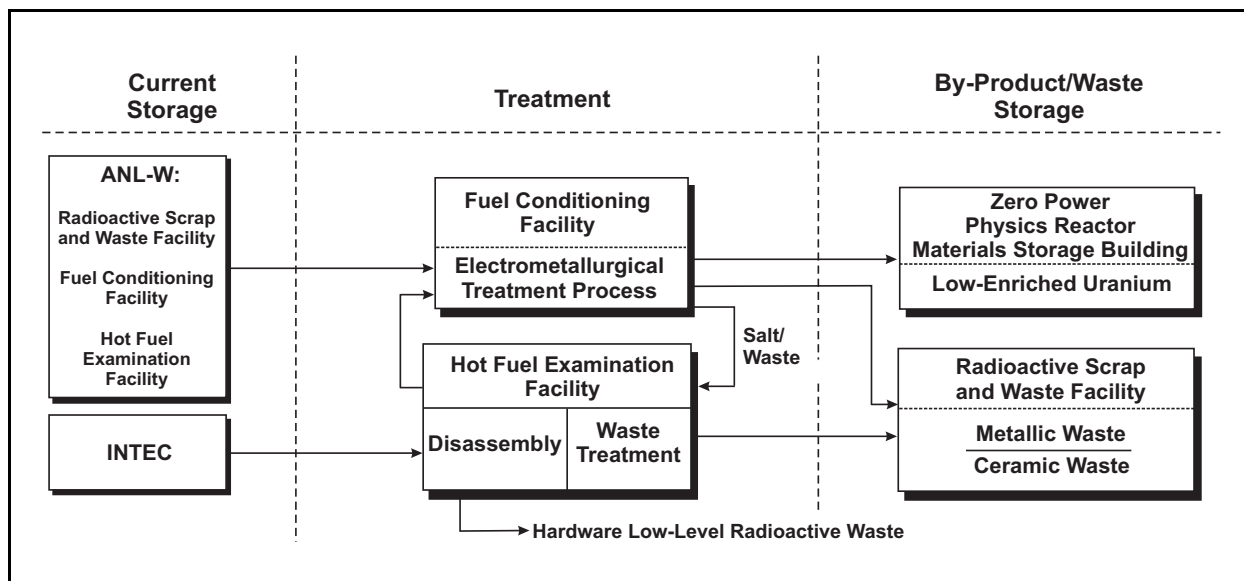


Figure 2–23 Schematic for Driver and Blanket Spent Nuclear Fuel Treatment in Alternative 1

2.5.3 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be packaged in high-integrity stainless steel cans at ANL-W after the sodium has been removed without decladding, as discussed in Section 2.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel would take place at the Hot Fuel Examination Facility at ANL-W. The packaging in high-integrity cans would take place in the same facility. The high-integrity cans would be transferred to the Radioactive Scrap and Waste Facility for storage. **Figure 2–24** illustrates the steps of the process for the blanket sodium-bonded spent nuclear fuel.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated using the electrometallurgical treatment process as described in Section 2.5.2 for Alternative 1.

Implementing this alternative at the Hot Fuel Examination Facility would require the installation of equipment for sodium removal activities. In addition, some new waste handling equipment would be needed for the electrometallurgical treatment of the driver sodium-bonded spent nuclear fuel.

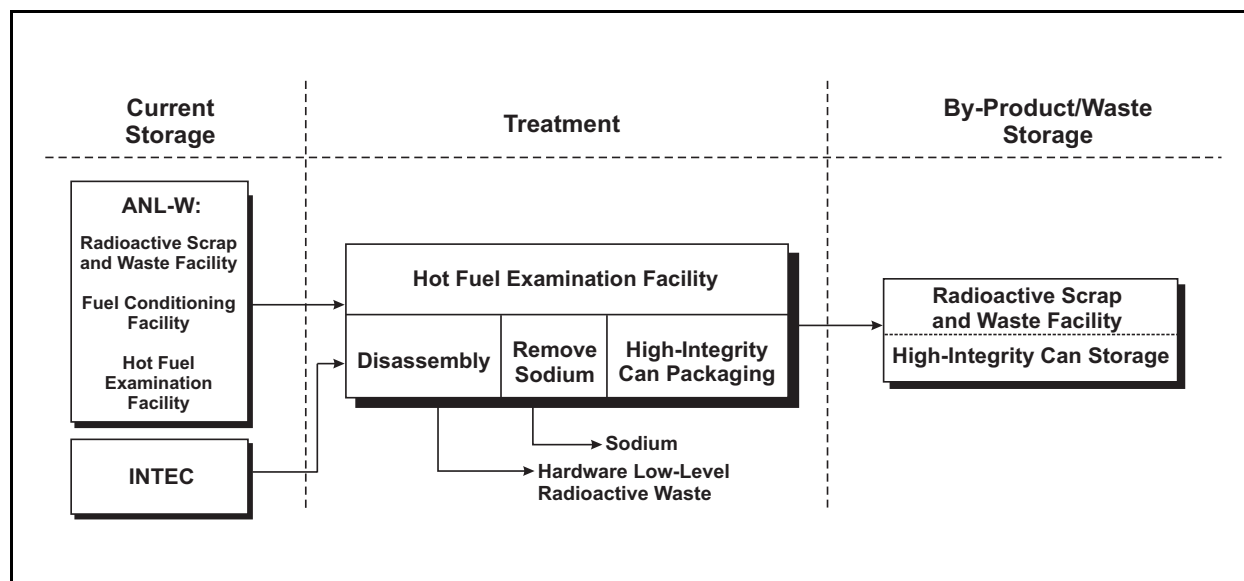


Figure 2-24 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 2

Packaging the blanket spent nuclear fuel in high-integrity cans could start by approximately 2003. It would take approximately six years to complete. Electrometallurgical treatment of the driver spent nuclear fuel would start in 2000 and would be completed in approximately seven years.

2.5.4 Alternative 3: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; PUREX Process Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged in aluminum cans and shipped to SRS for treatment using the PUREX process at the SRS F-Canyon facility. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W, as discussed in Section 2.3.9.

The decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal would take place at the Hot Fuel Examination Facility at ANL-W. Equipment for decladding and sodium removal would need to be installed for this purpose. After decladding and sodium removal, the blanket spent nuclear fuel pins would be packaged and stored temporarily at the Hot Fuel Examination Facility to await shipment to SRS.

At SRS, the cans containing blanket spent nuclear fuel pins would be unpacked at the F-Canyon facility before treatment using the PUREX process. No modifications to that facility would be needed. Waste from the process containing the fission products and transuranic isotopes other than plutonium would be transferred to the Defense Waste Processing Facility where it would be converted to borosilicate glass logs and stored pending ultimate disposal. Approximately 260 kilograms (572 pounds) of separated plutonium would be immobilized using the can-in-canister technology at SRS for eventual geologic repository disposal in accordance with the Record of Decision (75 FR 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999c). Depleted uranium would be transferred to a storage yard for depleted uranium at the site. **Figure 2-25** illustrates the process steps for the blanket spent nuclear fuel at ANL-W and SRS.

Considering the commitment of F-Canyon to other DOE missions, PUREX processing of the blanket spent nuclear fuel would start no earlier than 2005, and would last less than one year. Decladding and sodium

removal activities at ANL-W would not start earlier than 2003. Therefore, these activities would determine the length of the process.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment processes described in Section 2.5.2 for Alternative 1. As in the case of Alternative 2, electrometallurgical treatment of the driver spent nuclear fuel could start in 2000 and could be completed in approximately seven years. The process steps for the treatment of the driver spent nuclear fuel are shown in Figure 2-23.

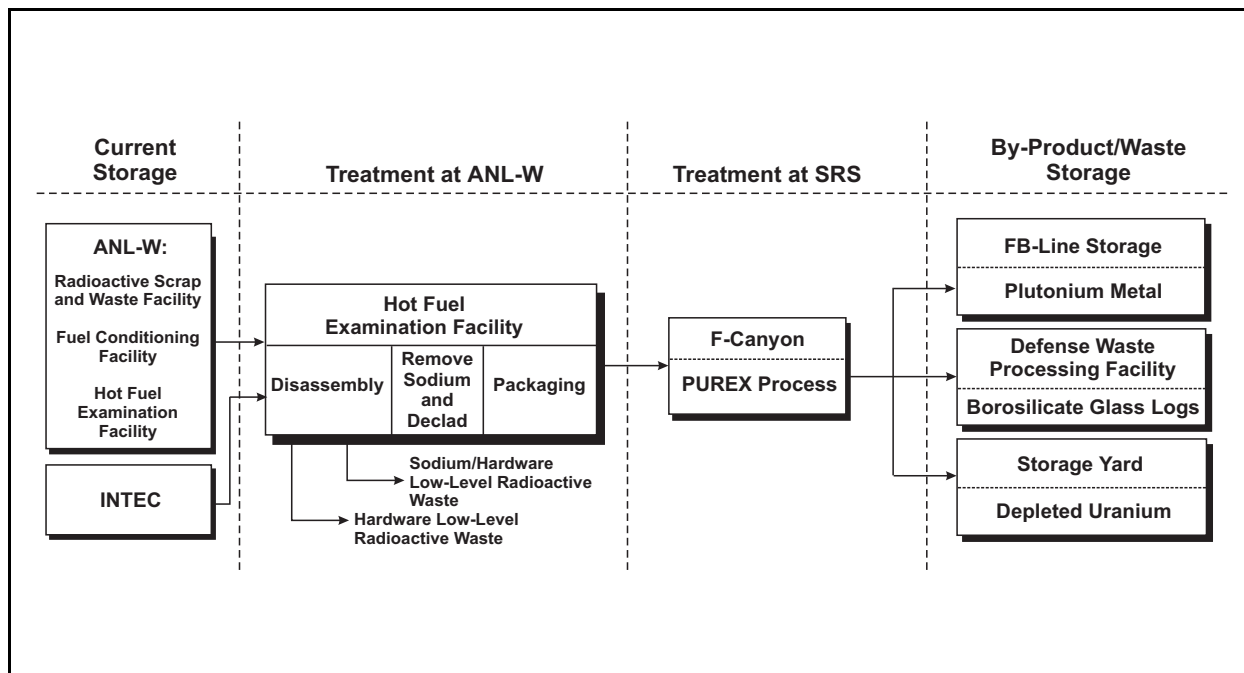


Figure 2-25 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 3

2.5.5 Alternative 4: Melt and Dilute Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the blanket spent nuclear fuel elements (approximately 57 metric tons of heavy metal) would be treated at ANL-W using the melt and dilute Option 2 process described in Section 2.3.4. Prior to treatment, the metallic sodium would be removed without decladding at ANL-W, as discussed in Section 2.3.9. Removal of the sodium from the sodium-bonded blanket spent nuclear fuel could take place at the Hot Fuel Examination Facility at ANL-W. Equipment for sodium removal would need to be installed. Equipment necessary for the melt and dilute process would need to be installed at the Hot Fuel Examination Facility, including the addition of the melter and an off-gas system.

Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be transferred to the Radioactive Scrap and Waste Facility for storage pending ultimate disposal. **Figure 2-26** illustrates the process steps for the sodium-bonded blanket spent nuclear fuel.

Treatment of blanket spent nuclear fuel at ANL-W using the melt and dilute process could start as early as 2005 and could be completed in eight years.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section 2.5.2 for Alternative 1. Treatment of the driver spent nuclear fuel could start as early as 2000 and could be completed in approximately seven years. The process steps for the treatment of the driver sodium-bonded spent nuclear fuel are shown in Figure 2–23.

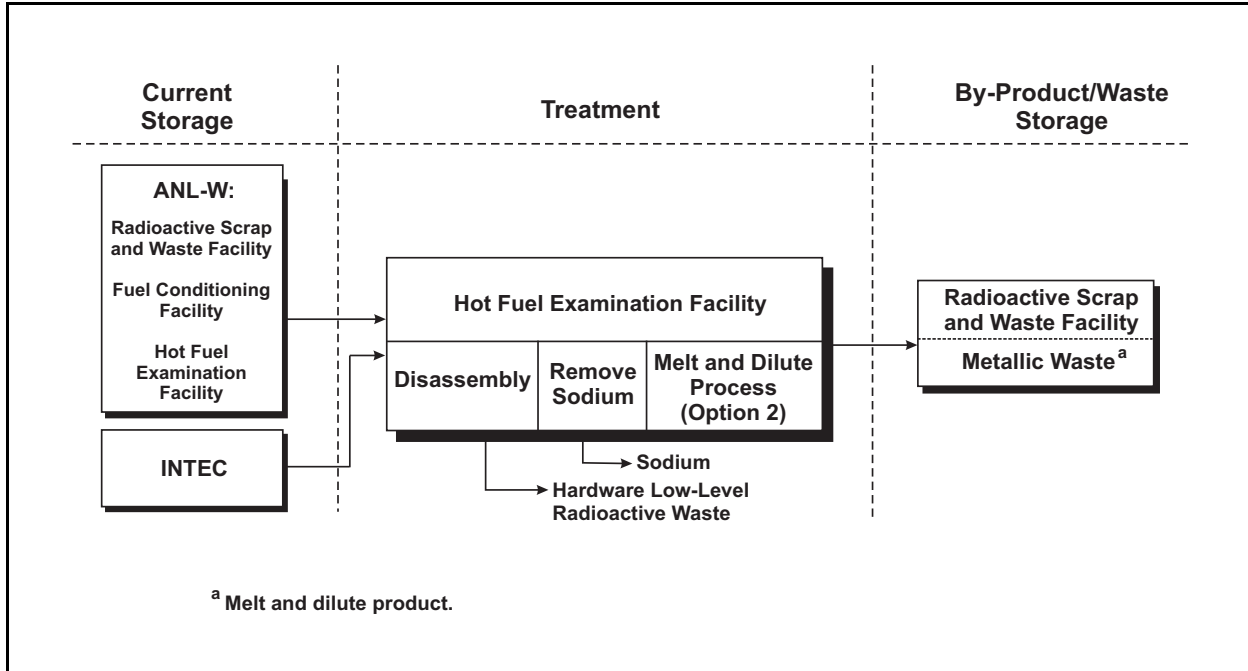


Figure 2–26 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 4

2.5.6 Alternative 5: Declad and Clean Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W; Melt and Dilute Blanket Fuel at SRS

Under this alternative, the blanket spent nuclear fuel pins (approximately 57 metric tons of heavy metal) would be packaged and shipped to SRS for treatment. The blanket spent nuclear fuel pins would be separated from the cladding and cleaned to remove the metallic sodium at ANL-W. The declad and cleaned blanket spent nuclear fuel pins would be received at Building 105-L at SRS and treated using the melt and dilute Option 1 process, as described in Section 2.3.4.

Decladding of the sodium-bonded blanket spent nuclear fuel and sodium removal would take place at the Hot Fuel Examination Facility at ANL-W, as discussed in Section 2.3.9. Spent nuclear fuel currently stored at ANL-W facilities could be transported directly to the Hot Fuel Examination Facility. After decladding and sodium removal, the blanket spent nuclear fuel pins would be packaged and stored temporarily at the Hot Fuel Examination Facility pending shipment to SRS.

At SRS, the cans containing the blanket spent nuclear fuel pins would be unpacked at Building 105-L and the blanket spent nuclear fuel pins would be treated using the melt and dilute process. For the purpose of evaluating this alternative, it is assumed that the melt and dilute facility is operational at SRS, as proposed in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000).

Metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be stored at the L-Area storage pending ultimate disposal.

Figure 2–27 illustrates the process steps for the blanket spent nuclear fuel at ANL-W and SRS.

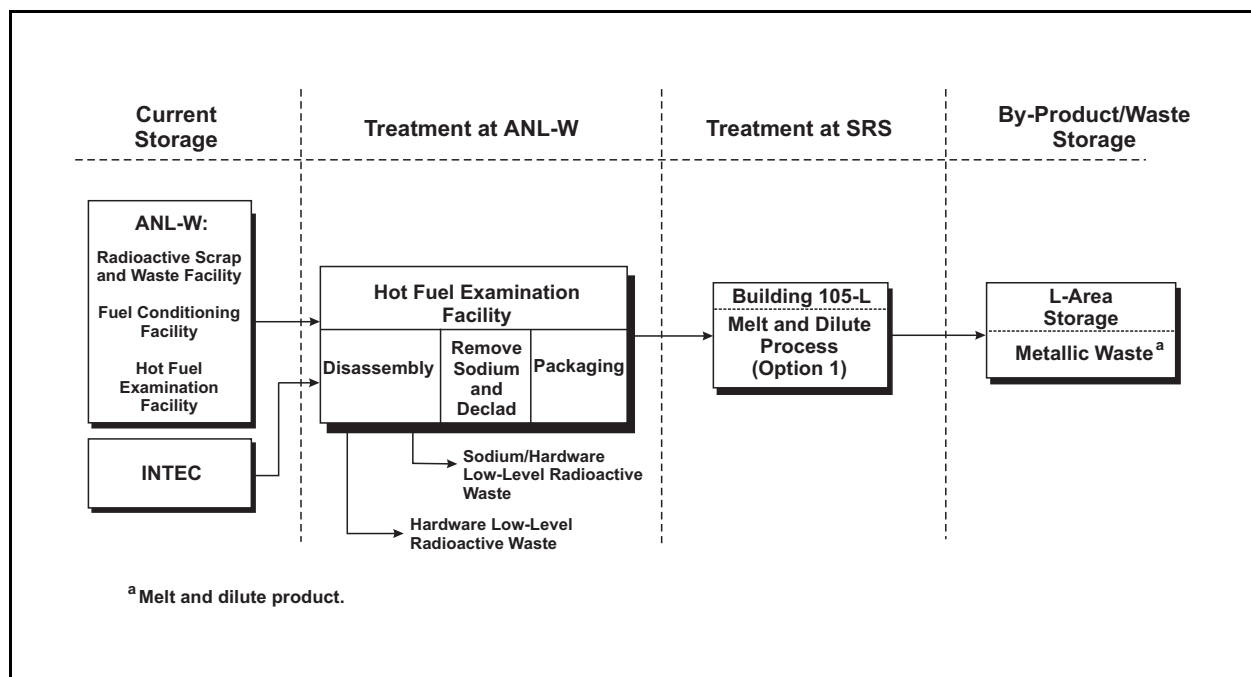


Figure 2–27 Schematic for Blanket Spent Nuclear Fuel Treatment in Alternative 5

Treatment of the blanket spent nuclear fuel at SRS would start around 2035. The facility would be operational in 2005 and is committed to other DOE missions until 2035. If additional capacity becomes available, treatment could start as soon as 2020. The treatment process would last approximately three years. Until 2035, there would be ample time for blanket spent nuclear fuel decladding and sodium removal activities at ANL-W.

The sodium-bonded driver spent nuclear fuel (approximately 3 metric tons of heavy metal) would be treated at ANL-W using the electrometallurgical treatment process described in Section 2.5.2 for Alternative 1. Treatment of the driver spent nuclear fuel at ANL-W could start in 2000 and could be completed in approximately seven years. The process steps for the treatment of the driver sodium-bonded spent nuclear fuel are shown in Figure 2–23.

2.5.7 Alternative 6: Melt and Dilute Blanket and Driver Fuel at ANL-W

Under this alternative, both the sodium-bonded blanket and driver spent nuclear fuel would be treated in the Hot Fuel Examination Facility at ANL-W using Options 2 and 3 of the melt and dilute process discussed in Section 2.3.4. Option 2 would be used for the blanket spent nuclear fuel, and Option 3 would be used for the driver spent nuclear fuel except for 0.1 metric tons (0.11 tons) of oxide, carbide, and nitride fuel, which would not be treated under the alternative. **Figure 2–28** illustrates the steps for the alternative.

Removal of the sodium from the blanket spent nuclear fuel and, to the extent practical, from the driver spent nuclear fuel would take place at the Hot Fuel Examination Facility. Equipment for sodium removal activities and the melt and dilute process would need to be installed in the inert cell of the facility.

The metallic waste resulting from the melt and dilute process containing fission products, depleted uranium, and transuranic elements would be transferred to the Radioactive Scrap and Waste Facility for storage pending ultimate disposal.

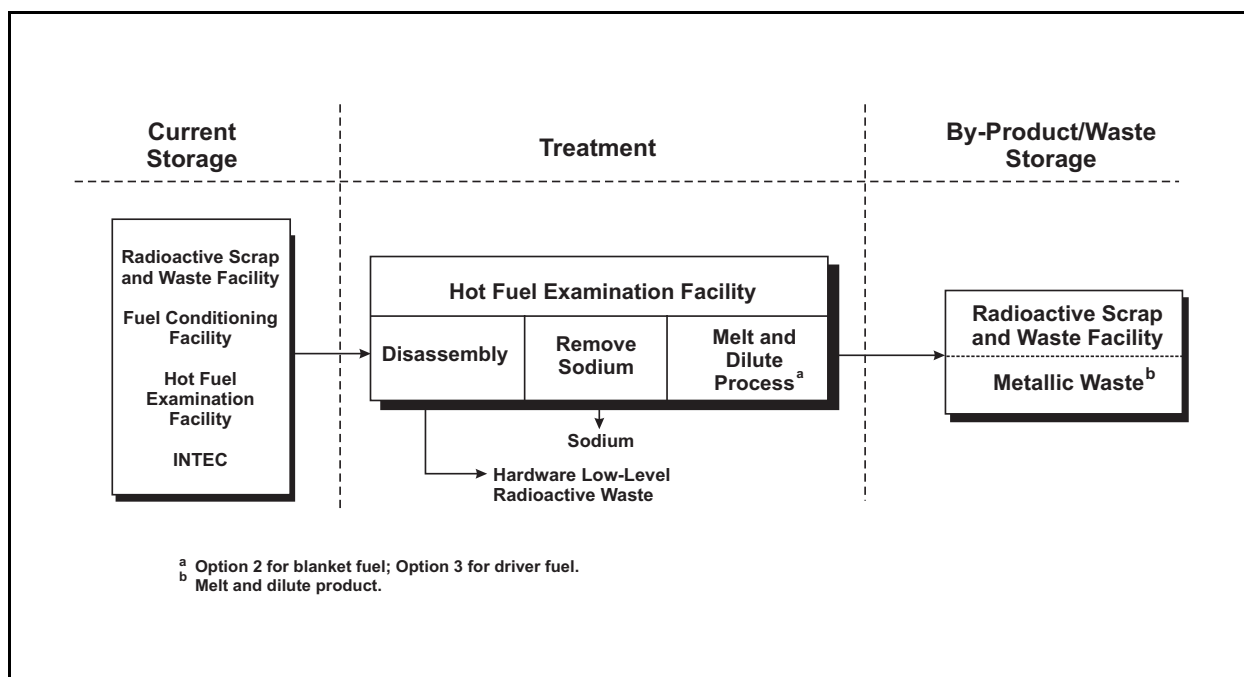


Figure 2–28 Schematic for Driver and Blanket Spent Nuclear Fuel in Alternative 6

The melt and dilute process at ANL-W could start as early as 2003 and would take approximately 12 years to complete for all driver and blanket sodium-bonded spent nuclear fuel.

2.6 ALTERNATIVES CONSIDERED AND DISMISSED

In identifying the reasonable alternatives for evaluation in this EIS, two separate issues led to the determination of alternatives that were considered and dismissed: (1) the level of maturity of the alternative technologies; and (2) the level of effort required to modify an existing facility to implement a specific technology. The construction of new facilities when existing facilities are still operative was not considered a reasonable option because of impacts and cost implications. Among the treatment technologies discussed in Section 2.3, the GMODS process and the direct plasma arc-vitreous ceramic process are not as mature as the electrometallurgical, melt and dilute, and PUREX processes when applied to sodium-bonded spent nuclear fuel. The GMODS and plasma arc processes both require significant and extensive research and development before they can be successfully proven to treat sodium-bonded spent nuclear fuel. The GMODS and plasma arc-vitreous ceramic processes each present specific technological challenges that cannot be answered without the construction, operation, and considerable engineering analysis of pilot-scale plants. In comparison, the melt and dilute process is being tested and evaluated, and has been selected as the Preferred Alternative for treatment of almost all aluminum-based spent nuclear fuel at SRS (DOE 2000). Use of the melt and dilute process for sodium-bonded driver spent nuclear fuel only requires technology enhancement, which DOE already has proposed for treating other spent nuclear fuel. In addition, unlike the other technologies that would require no new construction, GMODS and plasma arc processes would require the installation of large, specialized equipment in new hot cell facilities, the size and complexity of which are not sufficiently determined to allow detailed environmental impact analysis.

GMODS Process

The GMODS process, although similar to the melt and dilute process because of its thermal treatment, has not been developed beyond the laboratory scale. Several developmental steps would be required before it could be deemed a mature process. These include: detailed process development, resolution of containment concerns, testing, and a pilot plant demonstration to address technology risks (e.g., reliability and throughput).

GMODS would require large, specialized equipment to be installed in eight new large hot cell facilities. GMODS would dissolve the fuel elements or fuel assemblies entirely in a lead/lead-oxide system. An off-gas treatment system similar to that for the melt and dilute process would be required to treat the radioactive elements volatilized at about 1,000 °C (1,830 °F). The GMODS equipment could produce an intermediate waste form containing most of the actinides, fission products, and structural materials. After some preprocessing, the waste stream would be fed into the melter for the production of a new type of borosilicate glass log. These logs would contain uranium, other actinides, and structural elements in addition to the fission products. Because of the highly corrosive nature of the chemicals in the system, the technical feasibility of the alternative has not been established. This would add an additional degree of uncertainty to the waste estimates, as well as to the ultimate success of the fuel conditioning project.

Direct Plasma Arc-Vitreous Ceramic Process

The direct plasma arc-vitreous ceramic process is being used for the vitrification of low-level mixed waste. However, vitrification of spent nuclear fuel by this process is understood only on a conceptual level. The plasma arc treatment method would require large, complex equipment to be installed in a new, specially constructed hot cell facility. Such a facility could be constructed next to the Hot Fuel Examination Facility at ANL-W to secure some services. It would require the installation of equipment to cut the fuel assemblies into small pieces, a ceramic melter (furnace) to melt and oxidize the pieces at temperatures at least as high as 1,600 °C (2,900 °F), and an off-gas treatment system. As with the GMODS and melt and dilute processes, uranium and plutonium are not separated during the process. The conditioned spent nuclear fuel form would be vitreous ceramic and would include the sodium in a stable form. As with all processes that dissolve or melt spent nuclear fuel, the plasma arc process would produce radioactive off-gases. These gases would be filtered and treated, and the filter and treatment media would be stabilized into an acceptable waste form by a yet-to-be-determined process. The process would require testing in a pilot-scale plant to address the reliability of the plasma system.

The high temperatures of the process could increase the radioactive materials available for release during normal operation and accident conditions, thus increasing the exposure risk to members of the general public. Compared to other alternatives, there is a substantial uncertainty about the risk from accident conditions, considering the complexity of the off-gas treatment system. Because of the high temperature, more radioactive elements would be volatilized. In addition, considerable development would be required to produce very high- temperature rotating equipment that would operate reliably in a hot cell environment.

Chloride Volatility Process

The chloride volatility process design is in an early conceptual stage. The process needs high temperatures and chlorination for volatilization and chemical reactions to separate various fission products from uranium. This treatment technology would require a very elaborate gaseous separation process with potentially significant occupational and public risk in comparison to other treatment technologies, from both the volatilized fission products and the chlorine gas.

Electrometallurgical Treatment at INEEL Test Area North

Treatment of sodium-bonded spent nuclear fuel using the electrometallurgical treatment process at INEEL's Test Area North was considered and dismissed because Test Area North would require extensive modification to treat sodium-bonded spent nuclear fuel. Implementation of this alternative would require the construction of an argon hot cell. In addition, it would require either the procurement of new equipment or the transfer of already-contaminated equipment and other systems existing at ANL-W.

Treatment of Driver or Clad Blanket Spent Nuclear Fuel Using SRS PUREX Process

As discussed in Section 2.3.2, use of the PUREX process facilities at SRS for the treatment of sodium-bonded spent nuclear fuel would require the development and installation of a versatile front-end process to handle mechanical decladding, sodium removal, and zirconium sludge formation for EBR-II spent nuclear fuel. Such development does not appear justified for the sole purpose of treating the relatively small quantity of driver spent nuclear fuel.

Treatment of Driver Spent Nuclear Fuel Using SRS Melt and Dilute Process

As discussed in Section 2.3.4, the treatment of driver spent nuclear fuel would require a modified melt and dilute process that would be capable of handling the sodium volatilized from processing chopped driver spent nuclear fuel elements with the cladding intact. To accomplish this at SRS, significant design changes would be required from the process that DOE has proposed for the aluminum-clad spent nuclear fuel, which does not contain sodium. These design changes do not appear justified for the sole purpose of treating the relatively small quantity of driver spent nuclear fuel.

Treatment Using INEEL PUREX Process

Sodium-bonded spent nuclear fuel from EBR-II was being processed at the Idaho Chemical Processing Plant (now INTEC) using a PUREX process. DOE stopped processing at INTEC as a matter of policy in 1992, and the facility was permanently shut down. Reactivation of the facility is not practical and the alternative was dismissed.

2.7 ULTIMATE DISPOSITION

One of the technical risks in implementing any of the sodium-bonded spent nuclear fuel treatment methods is the uncertainty surrounding the acceptability of DOE spent nuclear fuel for placement in a potential repository. DOE would receive a license from the NRC to receive and store spent nuclear fuel in a repository (10 CFR 60 or draft 10 CFR 63). In order to obtain a license, DOE must develop acceptance criteria that establish the condition of the spent nuclear fuel for disposal and demonstrate that the criteria will meet NRC standards. Any spent nuclear fuel packaging or treatment technology must be capable of putting fuel in a form that will satisfy the acceptance criteria requirements. DOE's Office of Civilian Radioactive Waste Management has responsibility for a Federal repository. It is working to refine its acceptance criteria to ensure that spent nuclear fuel and high-level radioactive waste are packaged suitably for disposal. DOE has drafted preliminary acceptance criteria that are being used to assess the feasibility of DOE spent nuclear fuel disposition options (DOE 1999a). If the repository is developed, final acceptance criteria will not be available until after the NRC issues its construction authorization, based on the successful demonstration of safe, long-term performance of the candidate repository in accordance with NRC regulations. Until such time, the preliminary acceptance criteria tend to be conservative to allow for uncertainties in performance of engineered and natural barriers and how such performance will impact public and worker health and safety, and material isolation.

To ensure that the treatment option DOE selects will result in a product that is likely to meet the acceptance criteria, DOE is working with the NRC to obtain comments on the research and development work that DOE will perform to establish treatment technology specifications. To provide additional independent evaluation of the suitability of new treatment technologies, DOE requested that the National Academy of Sciences' National Research Council provide recommendations regarding DOE's sodium-bonded spent nuclear fuel treatment and disposition program. In a recent report (NAS 1998), the National Research Council recommended that the Electrometallurgical Treatment Research and Demonstration Project be carried to completion. The Council also expressed the opinion that, with the exception of the PUREX process, all other alternatives to the electrometallurgical process were at an early stage of development.

2.8 PREFERRED ALTERNATIVE

Council on Environmental Quality regulations (40 CFR 1502.14e) require that an agency identify its preferred alternative(s) in the final EIS. After careful consideration of public comments and programmatic, environmental, nonproliferation, and cost issues, DOE has identified electrometallurgical treatment as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel. Thus, the Preferred Alternative is a combination of Alternative 1 and the No Action Alternative. This combination would result in 26 metric tons of heavy metal of EBR-II and miscellaneous spent nuclear fuel being treated using the electrometallurgical process and 34 metric tons of heavy metal of Fermi-1 blanket spent nuclear fuel remaining in storage, pending a subsequent decision on its long-term management. The environmental consequences of the Preferred Alternative are addressed in Section 4.10.

DOE will validate the cost of using alternative treatment techniques (e.g., sodium removal and placement in high-integrity cans) for the Fermi-1 blanket spent nuclear fuel. These techniques may be economically favorable for the Fermi-1 blanket spent nuclear fuel because of characteristics that distinguish it from the EBR-II spent nuclear fuel. For example, the Fermi-1 blanket spent nuclear fuel does not require the extensive safeguards and security measures that are required for the EBR-II blanket fuel. The difference in security requirements for these two types of fuel is a result of the difference in plutonium content. The EBR-II blanket fuel has 30 times more plutonium at a greater concentration than the Fermi-1 blanket fuel.

Should DOE select the Preferred Alternative in the Record of Decision, DOE would proceed with the electrometallurgical treatment of the EBR-II sodium-bonded spent nuclear fuel and monitor the results and costs while continuing the development of sodium removal techniques for the Fermi-1 blanket spent nuclear fuel. Sodium removal would increase the number of long-term management options for the Fermi-1 fuel. While EBR-II spent nuclear fuel is undergoing electrometallurgical treatment and the Fermi-1 spent nuclear fuel remains in storage, DOE has approximately four years in which to evaluate the operating experience of electrometallurgical treatment technology and further develop other alternatives for the Fermi-1 spent nuclear fuel. After these data are evaluated, DOE would decide whether to treat the Fermi-1 blanket spent nuclear fuel using electrometallurgical treatment or to use another treatment method and/or disposal technique.

Before making a decision to treat or dispose of the Fermi-1 blanket spent nuclear fuel, DOE will determine whether the analysis in this EIS is adequate to support a subsequent Record of Decision or whether additional NEPA review is required. In any case, DOE will notify the public of its preferred approach for the Fermi-1 blanket spent nuclear fuel at least 30 days before issuing a Record of Decision regarding treatment or disposal.

For several years, DOE has been actively developing electrometallurgical treatment technology specifically for the management of sodium-bonded spent nuclear fuel. Having completed a successful demonstration of electrometallurgical treatment, DOE believes that this technology has the highest probability of meeting the Department's needs for managing much of the sodium-bonded spent nuclear fuel. Electrometallurgical

technology would convert the reactive fuel into ceramic and metallic waste forms, both of which are more stable than untreated sodium-bonded spent nuclear fuel. In addition, uranium would be separated from the spent nuclear fuel, blended with depleted uranium if needed to lower enrichment levels, and cast into ingots to be stored until a disposition decision is made through a separate NEPA review. Most of the plutonium would be disposed of in the ceramic waste form, with the remaining small fraction disposed of in the metallic waste form. Currently, the only waste form that has been tested and analyzed extensively under geologic repository conditions and may be accepted for repository disposal is borosilicate glass. Tests have shown the ceramic and metallic waste forms from electrometallurgical treatment may perform as well as the standard borosilicate glass waste form. The ceramic and metallic waste forms would require less storage volume than untreated spent nuclear fuel.

2.9 SUMMARY OF ENVIRONMENTAL IMPACTS

This section summarizes the environmental impacts associated with the No Action Alternative and the six reasonable alternatives under the proposed action that are evaluated in detail as part of this EIS (see Section 2.5). The information presented in this section is based on Chapter 4, which provides a detailed discussion of the impacts on potentially affected environmental areas. Such environmental areas include: air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, waste management, and transportation.

For the alternatives evaluated, the analyses showed that there would be no significant impacts on air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, and transportation. The radiological and nonradiological gas and liquid releases, as well as the associated exposures to workers and the public, would be well below regulatory standards and guidelines and no mitigation measures would be warranted. In addition, the environmental impact analysis indicates that there are no significant impacts that would discriminate one alternative over another.

The only significant difference between the No Action Alternative and the reasonable alternatives is in the area of waste generation. All of the proposed alternatives result in a decrease in high-level radioactive waste volume as compared to the direct disposal volume associated with the No Action Alternative.

For the No Action Alternative and the six alternatives evaluated, the proposed facilities already exist. Except for internal building modifications and new equipment installation, no construction activities would be required. Therefore, DOE has determined the proposed action would have minimal or no impacts on land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources. Impacts to these resources were not evaluated in detail in the EIS.

The impacts of the No Action Alternative are presented first as a baseline for comparing the impacts under the proposed action. A summary of the environmental impacts for the No Action Alternative and the other six reasonable alternatives is presented as **Table 2-4**. Section 4.1, Chapter 4, provides information on the categories and results illustrated in Table 2-4.

2.9.1 No Action Alternative Impacts

Under the No Action Alternative, the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed from the interior of the fuel elements). The EIS evaluates the impacts of two separate options under this alternative:

- a. Monitoring and stabilizing the sodium-bonded spent nuclear fuel as necessary for continued storage at current locations until 2035, or until a new treatment technology (such as GMODS or plasma arc) is developed.

- b. Direct disposal of sodium-bonded spent nuclear fuel in a geologic repository by packaging the fuel in high-integrity cans. As discussed in Section 2.3.8, the direct disposal option may not meet current NRC and/or RCRA requirements.

The activities associated with the preparation of sodium-bonded spent nuclear fuel for direct disposal would be similar to those needed to prepare the fuel for interim or continued storage. Both require that fuel be transferred to a hot cell, examined (nondestructive examination) and characterized, and repackaged. The only difference between these two options is that for direct disposal, the sodium-bonded spent nuclear fuel would be placed in high-integrity cans in preparation for ultimate disposal, while for storage it would not be placed in high-integrity cans. Direct disposal also requires consideration of criticality safety, thereby limiting the amount of driver spent nuclear fuel that could be packaged in a canister, leading to higher repository volume requirements.

Air Quality

For both options under the No Action Alternative, activities at ANL-W and INTEC would have a negligible impact on existing air quality. Radiological emissions would also be low and well below regulatory concern. Approximately 810 curies would be released over a 35-year period from possible fuel degradation during storage. Air quality for INEEL is not expected to change as a result of the No Action Alternative.

Water Resources

Surface water is not used at ANL-W and INTEC and this would not change under either option of the No Action Alternative. Groundwater use, primarily domestic consumption, could decrease if there is a reduction in workers at ANL-W.

No changes are expected in liquid effluent discharges. There are currently no discharges to surface waters (radiological or nonradiological) except for discharges of nonhazardous liquid waste, which are monitored and subject to National Pollutant Discharge Elimination System (NPDES) permit requirements.

Socioeconomics

Under either option of the No Action Alternative, there could be a reduction of approximately 350 workers at ANL-W. This reduction could result in the loss of 940 additional indirect jobs in the economic region. The reduction would take place over time, therefore, the No Action Alternative would not result in any noticeable changes in the existing regional economy, housing characteristics, or community services.

Public and Occupational Health and Safety

The risk to the health and safety of workers and the public under either option of the No Action Alternative would be from the potential exposure to radiological or hazardous chemical emissions during normal operation or accident conditions.

Radiological Exposures

Routine radioactive releases associated with either option of the No Action Alternative at ANL-W and INTEC would be small. The maximum annual dose to the population within 80 kilometers (50 miles) from these releases would be 0.0015 person-rem. This exposure would lead to 7.5×10^{-7} additional latent cancer fatalities in the population, or one chance in 1.3 million that the exposed population would experience a latent cancer fatality. For comparison purposes, the collective dose for the same population in the year 2010 from natural background radiation would be 86,250 person-rem. The maximally exposed offsite individual

would receive a maximum of 0.00026 millirem, and the risk of developing a fatal cancer from this exposure would be 1.3×10^{-10} , or one chance in 7.7 billion. The average worker would receive 60 millirem per year, and the risk of developing a cancer from this exposure would be 0.000024, or one chance in 41,666.

Similarly, for the 35-year duration of the project, the population dose from routine releases would be 0.013 person-rem (with a latent cancer fatality risk of 6.5×10^{-6}); the maximally exposed offsite individual dose would be 0.0023 millirem (with a latent cancer fatality risk of 1.1×10^{-9}); and the worker population dose would be 209 person-rem (with a latent cancer fatality risk of 0.084).

The maximum annual cancer risk from postulated accident conditions under continued storage or direct disposal of the No Action Alternative at ANL-W would be 5.6×10^{-6} for the population within 80 kilometers (50 miles). The annual cancer risk for the maximally exposed offsite individual would be 4.8×10^{-8} , and 1.5×10^{-8} for the noninvolved worker.

Hazardous Chemical Exposures

Hazardous chemical impacts resulting from either option of the No Action Alternative would be small because any emissions of hazardous chemicals from activities under the No Action Alternative would be very low.

Hazardous chemical impacts under accident conditions, evaluated in terms of Emergency Response Planning Guideline (ERPG) values, indicate that under either option of the No Action Alternative, the worst postulated accident conditions would result in less than ERPG-1 conditions for a worker or the maximally exposed offsite individual.

Environmental Justice

As discussed above, the impacts from either option of the No Action Alternative on the health and safety of the public would be very small regardless of the racial and ethnic composition of the population and independent of the economic status of the individuals comprising the population in 2010.

Waste Management

For both options under the No Action Alternative, various types of waste would continue to be generated at ANL-W. These include low-level radioactive, transuranic, mixed, hazardous, and nonhazardous waste. These waste types are associated with the operation of the facilities where the sodium-bonded spent nuclear fuel is stored. High-level radioactive waste in metallic and ceramic forms generated as a result of completing the Electrometallurgical Treatment Research and Demonstration Project waste processing would be stored at the Radioactive Scrap and Waste Facility pending disposal. Finally, some additional low-level radioactive waste and transuranic waste would be generated from the deactivation of the demonstration project. The volumes of these waste types are presented in Table 2-4.

Transportation

The No Action Alternative involves the movement of sodium-bonded spent nuclear fuel within the INEEL site. All fuel stored at INTEC would be moved to ANL-W for repackaging in preparation for transport either to a geologic repository or out of the State of Idaho by 2035.

The dose to transportation workers from all transportation activities under the No Action Alternative is estimated at 0.003 person-rem; the dose to the public would be 0.022 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.2×10^{-6} latent cancer fatalities among transportation

workers and 0.000011 latent cancer fatalities in the total affected population over the duration of the transportation activities.

The dose to the affected population from postulated accidents during transportation would be less than 1.0×10^{-6} person-rem, resulting in less than 1.0×10^{-9} latent cancer fatalities. Nonradiological transportation accidents are estimated to result in 0.00012 traffic fatalities.

2.9.2 Proposed Action Impacts

Under the proposed action, the EIS evaluates six distinct alternatives, as described in Section 2.5 and illustrated in Figure 2-22. Alternative 1 proposes to treat both driver and blanket spent nuclear fuel using the electrometallurgical method at ANL-W. Alternatives 2 through 5 propose to treat the driver spent nuclear fuel using the electrometallurgical method (as in Alternative 1), but other methods and/or sites would be used for the blanket spent nuclear fuel, including: the high-integrity can packaging at ANL-W (Alternative 2); the PUREX process at SRS (Alternative 3); the melt and dilute process at ANL-W (Alternative 4); and the melt and dilute process at SRS (Alternative 5). Alternative 6 proposes to treat both driver and blanket spent nuclear fuel using the melt and dilute method at ANL-W.

All alternatives under the proposed action would have very small impacts on air quality, water resources, socioeconomics, public and occupational health and safety, environmental justice, and transportation areas of the environment in and around the INEEL/ANL-W and SRS locations. For all alternatives, the radiological and nonradiological gaseous emissions and liquid effluent, as well as the associated exposures to workers and the public, would be well below regulatory standards and guidelines. A major difference between the No Action and proposed action alternatives is in the area of waste generation. Since the acceptability of chemically reactive sodium in a high-level radioactive waste repository is a primary concern in this EIS, it is important to consider the volume of high-level radioactive waste for all the proposed action alternatives. All the proposed action alternatives would result in a decrease in high-level radioactive waste volume as compared to the direct disposal No Action Alternative. The reduction in high-level radioactive waste volume for each alternative would be: 47 percent (Alternative 1); 71 percent (Alternative 2); 84 percent (Alternative 3); 58 percent (Alternative 4); 37 percent (Alternative 5); and 43 percent (Alternative 6).

Air Quality

The proposed action would have a negligible impact on existing air quality at ANL-W and SRS for each of the alternatives. Air quality at ANL-W and SRS would not change as a result of the proposed action.

During the year of maximum releases, radiological gaseous emissions would be well below regulatory concerns for each of the alternatives. Radiological gaseous emissions at ANL-W would be in the range of 770 (Alternative 1) to 2,162 (Alternative 6) curies of elemental tritium and 11,600 (Alternative 1) to 32,650 (Alternative 6) curies of krypton-85.

During the year of maximum releases, radiological gaseous emissions at SRS would be 54 (Alternative 5) to 162 (Alternative 3) curies of elemental tritium and 396 (Alternative 5) to 1,187 (Alternative 3) curies of krypton-85.

Water Resources

Surface water is not used at ANL-W, and this would not change under any of the alternatives proposed for ANL-W. Groundwater use, primarily domestic consumption, would remain at current levels, as the work force would be expected to remain at current levels for all alternatives of the proposed action.

No changes would be expected in liquid effluent discharges from any of the alternatives at ANL-W. There are currently no discharges to surface waters (radiological or nonradiological) except for discharges of nonhazardous liquid waste to the industrial pond, which are monitored and are subject to NPDES permit requirements.

Potential radioactive liquid effluent has been identified for the PUREX process at SRS under Alternative 3. Table 2–4 indicates some small quantities of tritium, and other isotopes. No radioactive liquid effluent has been identified for the melt and dilute process at SRS under Alternative 5.

Socioeconomics

All the alternatives under the proposed action assume that the treatment and management of the sodium-bonded spent nuclear fuel at ANL-W or SRS would not require an additional work force, but the activities would keep the work force from being reduced. Therefore, there would be no changes to the socioeconomic conditions in the vicinity of either ANL-W or SRS.

Public and Occupational Health and Safety

The potential risk of concern to the health and safety of the workers and the public under the proposed action would be from exposure to routine radiological emissions and hazardous chemical releases under normal operation or accident conditions. As indicated in Table 2–4, the risk would be small for all alternatives considered under the proposed action.

Radiological Exposures

Comparing alternatives at ANL-W, the maximum annual population dose from routine gaseous radioactive releases would range from 0.0028 person-rem (Alternative 1) to 0.012 person-rem (Alternative 6), with latent cancer fatalities in the range of 1.4×10^{-6} to 6.0×10^{-6} , respectively. The project total population dose would range from 0.016 person-rem (Alternative 1) to 0.024 person-rem (Alternative 6), with latent cancer fatalities in the range of 8.2×10^{-6} to 1.2×10^{-5} , respectively.

The maximum annual dose to the maximally exposed offsite individual at ANL-W would range from 0.00034 millirem (Alternative 1) to 0.002 millirem (Alternative 6), with a latent cancer fatality risk in the range of 1.7×10^{-10} to 1.0×10^{-9} , respectively. The project total dose to the maximally exposed offsite individual would range from 0.002 millirem (Alternative 1) to 0.004 (Alternative 6), with latent cancer fatality risks of 1.0×10^{-9} to 2.0×10^{-9} , respectively.

The collective annual dose to workers at ANL-W would be 22 person-rem for all alternatives. This corresponds to additional latent cancer fatalities of 0.0088. The average dose to a worker at ANL-W would be 60 millirem per year, which corresponds to a latent cancer fatality risk of 0.000024 per year. The project total worker dose would range from 231 person-rem (Alternatives 3 and 5) to 319 person-rem (Alternatives 1 and 4), with latent cancer fatalities ranging between 0.092 and 0.13, respectively.

Comparing alternatives at SRS, the maximum population dose from routine gaseous radioactive releases would range from 0.0076 person-rem per year for three years (Alternative 5) to 0.02 person-rem for the whole treatment period (Alternative 3), corresponding to additional latent cancer fatalities in the range of 0.000011 to 0.00001, respectively, for the whole treatment period.

The maximum dose to the maximally exposed offsite individual would range from 0.0001 millirem per year for three years (Alternative 5) to 0.00051 millirem (Alternative 3) for the whole treatment period, with latent cancer fatality risks of 1.5×10^{-10} and 2.6×10^{-10} , respectively, for the whole treatment period.

The maximum collective dose to workers at SRS would be 50 person-rem per year (Alternative 5) for three years. This corresponds to 0.06 additional latent cancer fatalities. The maximum average dose to a worker at SRS would be 500 millirem per year (Alternative 5), which corresponds to a latent cancer fatality risk of 0.0002 per year.

The highest annual latent cancer fatality risk for the population within 80 kilometers (50 miles) of ANL-W from postulated design-basis accident conditions under the proposed action would be 0.009 (Alternative 6, driver fuel, design-basis earthquake). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 0.000076 (Alternative 6, driver fuel, design-basis earthquake). The highest annual latent cancer fatality risk for the noninvolved worker would be 2.7×10^{-6} (Alternative 6, driver fuel, design-basis earthquake).

The highest annual latent cancer fatality risk for the population within 80 kilometers (50 miles) of ANL-W from postulated beyond-design-basis accident conditions under the proposed action would be 0.000013 (Alternatives 1 through 5, driver fuel, beyond-design-basis earthquake). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 2.2×10^{-7} (Alternatives 1 through 5, driver fuel, beyond-design-basis earthquake). The highest annual latent cancer fatality risk for the noninvolved worker would be 2.3×10^{-9} (Alternatives 1 through 5, blanket fuel, beyond-design-basis earthquake).

The highest annual latent cancer fatality risk for the population within 80 kilometers (50 miles) of SRS from postulated design-basis accident conditions under the proposed action would be 0.011 (Alternative 5, blanket fuel, loss of power). The highest annual latent cancer fatality risk for the maximally exposed offsite individual would be 6.6×10^{-6} (Alternative 5, blanket fuel, loss of power). The highest annual latent cancer fatality risk for the noninvolved worker would be 3.4×10^{-7} (Alternative 5, blanket fuel, loss of power).

Hazardous Chemical Exposures

Hazardous chemical impacts from normal operations for all alternatives under the proposed action would be small because the emissions of hazardous chemicals from the treatment and management of sodium-bonded spent nuclear fuel would be very low.

Hazardous chemical impacts under accident conditions, evaluated in terms of comparison to ERPG values, indicate that under the proposed action, all postulated hazardous chemical releases would not result in worse than ERPG-1 conditions for the noninvolved worker or the maximally exposed offsite individual at either ANL-W or SRS.

Environmental Justice

As discussed above, the impacts from the proposed action on the health and safety of the public would be very small, regardless of the racial and ethnic composition of the population and independent of the economic status of the individuals comprising the population in 2010.

Waste Management

Table 2-4 presents a comparison of the volumes of high-level radioactive, low-level radioactive, and transuranic waste generated by each of the alternatives. The alternatives would generate from 37 to 84 percent less high-level radioactive waste as compared to the direct disposal option of the No Action Alternative. Each of the alternatives would generate more transuranic waste, but Alternatives 1, 2, 4, and 6 would exceed this waste volume by a range of only 7 to 41 percent. Alternatives 3 and 5 would generate significantly greater volumes of transuranic waste, between 2.7 to 10 times the volume of transuranic waste generated by the direct disposal No Action Alternative.

All of the alternatives either would remove or convert the metallic sodium into a nonreactive form.

With respect to disposability and waste acceptance criteria, only the borosilicate glass waste form of Alternative 3 for blanket spent nuclear fuel has been tested and analyzed extensively under conditions relevant to a geologic repository. It is expected that other waste forms (e.g., ceramic, metallic, and possibly high-integrity cans not containing metallic sodium) would be suitable for repository disposal.

Transportation

The transportation activities for all alternatives under the proposed action would involve the movement of the sodium-bonded spent nuclear fuel within the INEEL site.

The incident-free dose to transportation workers from these activities would be between 0.0043 (Alternative 2) and 0.027 (Alternative 6) person-rem; the dose to the affected public would be between 0.032 (Alternative 2) and 0.2 (Alternative 6) person-rem. Accordingly, incident-free transportation activities would result in latent cancer fatalities in the range of 1.7×10^{-6} to 1.1×10^{-5} among transportation workers and 0.000016 to 0.0001 among the total affected population over the duration of the transportation activities.

The dose to the affected population from postulated accidents from these activities would be less than 1×10^{-6} person-rem, resulting in less than 1×10^{-9} latent cancer fatalities. Nonradiological traffic fatalities would be approximately 0.0001.

Transportation activities under Alternatives 3 and 5 would include, in addition, the movement of the blanket spent nuclear fuel pins from ANL-W to SRS. The incident-free dose to transportation workers from these activities would be 0.0012 person-rem; the dose to the public would be 0.012 person-rem. Accordingly, incident-free transportation activities would result in 4.7×10^{-7} latent cancer fatalities among transportation workers and 6.0×10^{-6} latent cancer fatalities in the total affected population over the duration of the transportation activities. Nonradiological fatalities from vehicle emissions during intersite transportation would be 0.00039 among affected urban populations along the transportation route.

The dose to the affected population from postulated accidents from these activities would be less than 3.0×10^{-6} person-rem, resulting in less than 1.5×10^{-9} additional latent cancer fatalities. Nonradiological traffic fatalities would be 0.0018.

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Table 2–4 Summary of Environmental Consequences for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel

Resource/Material Categories	No Action		Alternative 1		Alternative 2	
	ANL-W		ANL-W		ANL-W	
Air Quality - Radiological air emissions (curies per year)	Negligible impact 811 ^a		Negligible impact Tritium: 770 Krypton-85: 11,600		Negligible impact Tritium: 809 Krypton-85: 11,860	
Water Resources - Radiological liquid effluent (curies per year)	No impact No liquid effluent		No impact No liquid effluent		No impact No liquid effluent	
Socioeconomics	Loss of 350 direct jobs and 940 indirect jobs. No noticeable impact.		Work force maintained; No impact		Work force maintained; No impact	
Public and Occupational Health and Safety						
• Project duration	35 years		13 years		9 years	
• Normal operations (annual) ^c	Dose per year	LCF	Dose per year	LCF	Dose per year	LCF
- Population	0.0015	7.5×10^{-7}	0.0028	1.4×10^{-6}	0.003	1.5×10^{-6}
- MEI	0.00026	1.3×10^{-10}	0.00034	1.7×10^{-10}	0.00037	1.9×10^{-10}
- Average individual	6.2×10^{-6}	3.1×10^{-12}	0.000012	5.8×10^{-12}	0.000013	6.2×10^{-12}
- Worker population	22	0.0088	22	0.0088	22	0.0088
- Average worker	60	0.000024	60	0.000024	60	0.000024
• Normal operations (project total) ^c	Dose	LCF	Dose	LCF	Dose	LCF
- Population	0.013	6.5×10^{-6}	0.016	8.2×10^{-6}	0.017	8.3×10^{-6}
- MEI	0.0023	1.1×10^{-9}	0.002	1.0×10^{-9}	0.0021	1.0×10^{-9}
- Worker population	209	0.084	319	0.13	231	0.092
Hazardous chemicals						
- MEI	None		None		None	
• Accidents Maximum annual cancer risk						
- Population	5.6×10^{-6} (DBA) ^a		5.6×10^{-6} (DBA); 0.000013 (BDBA)		5.6×10^{-6} (DBA); 0.000013 (BDBA)	
- MEI	4.8×10^{-8} (DBA) ^a		4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)		4.8×10^{-8} (DBA); 2.2×10^{-7} (BDBA)	
- Noninvolved worker	1.5×10^{-8} (DBA) ^a		4.5×10^{-8} (DBA); 2.3×10^{-9} (BDBA)		1.5×10^{-8} (DBA); 1.5×10^{-9} (BDBA)	
Chemical accidents						
- MEI	Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
- Noninvolved worker	Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
Environmental Justice	No disproportionately high and adverse impact to minority or low-income populations					
Waste Management (cubic meters)						
• High-level radioactive waste	152 (Direct disposal volume) ^d		81.1		43.2 ^e	
• Low-level radioactive waste	904		862		733.7	
• Transuranic waste	12.1		14.1		10.7	
Transportation						
• Incident-free	Person-rem	LCF	Person-rem	LCF	Person-rem	LCF
- Population	0.022	0.000011	0.033	0.000016	0.032	0.000016
- Workers	0.003	1.2×10^{-6}	0.0044	1.8×10^{-6}	0.0043	1.7×10^{-6}
• Accidents						
- Population	less than 1.0×10^{-6}	less than 1.0×10^{-9}	less than 1.0×10^{-6}	less than 1.0×10^{-9}	less than 1.0×10^{-6}	less than 1.0×10^{-9}

ERPG = Emergency Response Planning Guideline; LCF = Latent Cancer Fatalities; MEI = Maximally Exposed Offsite Individual;

DBA = Design-Basis Accident; BDBA = Beyond-Design-Basis Accident

^a Represents total curies for 35 years; tritium: 51 curies; krypton-85: 760 curies; iodine-129: 0.000018 curies.

^b Over a period of six months.

Alternative 3				Alternative 4		Alternative 5				Alternative 6	
ANL-W		SRS ^b		ANL-W		ANL-W		SRS		ANL-W	
Negligible impact Tritium: 809 Krypton-85: 11,860		Negligible impact Tritium: 162 Krypton-85: 1,187		Negligible impact Tritium: 809 Krypton-85: 11,860		Negligible impact Tritium: 809 Krypton-85: 11,860		Negligible impact Tritium: 54 Krypton-85: 396		Negligible impact Tritium: 2,162 Krypton-85: 32,650	
No impact No liquid effluent		Negligible impact Tritium: 1.54 Other: less than 0.022		No impact No liquid effluent		No impact No liquid effluent		No impact No liquid effluent		No impact No liquid effluent	
Work force maintained; no impact		Work force maintained; no impact		Work force maintained; no impact		Work force maintained; no impact		Work force maintained; no impact		Work force maintained; no impact	
9 years		Less than 1 year		13 years		9 years		3 years		12 years	
Dose per year	LCF	Dose per year	LCF	Dose per year	LCF	Dose per year	LCF	Dose per year	LCF	Dose per year	LCF
0.003	1.5 × 10 ⁻⁶	0.02	0.000010	0.003	1.5 × 10 ⁻⁶	0.003	1.5 × 10 ⁻⁶	0.0076	3.8 × 10 ⁻⁶	0.012	6.1 × 10 ⁻⁶
0.00037	1.9 × 10 ⁻¹⁰	0.00051	2.6 × 10 ⁻¹⁰	0.00037	1.9 × 10 ⁻¹⁰	0.00037	1.9 × 10 ⁻¹⁰	0.00010	5.0 × 10 ⁻¹¹	0.002	1.0 × 10 ⁻⁹
0.000013	6.2 × 10 ⁻¹²	0.000024	1.2 × 10 ⁻¹¹	0.000013	6.2 × 10 ⁻¹²	0.000013	6.2 × 10 ⁻¹²	0.000011	5.5 × 10 ⁻¹²	0.000051	2.6 × 10 ⁻¹¹
22	0.0088	38	0.015	22	0.0088	22	0.0088	50	0.02	22	0.0088
60	0.000024	250	0.0001	60	0.000024	60	0.000024	500	0.0002	60	0.000024
Dose	LCF	Dose	LCF	Dose	LCF	Dose	LCF	Dose	LCF	Dose	LCF
0.017	8.3 × 10 ⁻⁶	0.02	0.00001	0.017	8.3 × 10 ⁻⁶	0.017	8.3 × 10 ⁻⁶	0.023	0.000011	0.024	0.000012
0.0021	1.0 × 10 ⁻⁹	0.00051	2.6 × 10 ⁻¹⁰	0.0021	1.0 × 10 ⁻⁹	0.0021	1.0 × 10 ⁻⁹	0.0003	1.5 × 10 ⁻¹⁰	0.004	2.0 × 10 ⁻⁹
231	0.092	38	0.015	319	0.13	231	0.092	150	0.06	297	0.12
None		None		None		None		None		None	
5.6 × 10 ⁻⁶ (DBA); 0.000013 (BDBA)		0.00014 (DBA)		0.00022 (DBA); 0.000013 (BDBA)		5.6 × 10 ⁻⁶ (DBA); 0.000013 (BDBA)		0.011		0.0090 (DBA)	
4.8 × 10 ⁻⁸ (DBA); 2.2 × 10 ⁻⁷ (BDBA)		7.2 × 10 ⁻⁸ (DBA)		1.9 × 10 ⁻⁶ (DBA); 2.2 × 10 ⁻⁷ (BDBA)		4.8 × 10 ⁻⁸ (DBA); 2.2 × 10 ⁻⁷ (BDBA)		6.6 × 10 ⁻⁶		0.000076 (DBA)	
1.5 × 10 ⁻⁸ (DBA); 1.5 × 10 ⁻⁹ (BDBA)		6.2 × 10 ⁻⁷ (DBA)		4.9 × 10 ⁻⁸ (DBA); 1.5 × 10 ⁻⁹ (BDBA)		1.5 × 10 ⁻⁸ (DBA); 1.5 × 10 ⁻⁹ (BDBA)		3.4 × 10 ⁻⁷		2.7 × 10 ⁻⁶ (DBA)	
Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1		Less than ERPG-1	
No disproportionately high and adverse impacts to minority or low-income populations											
23.6 (18 at ANL-W; 5.6 at SRS)				63.6		94.62 (18 at ANL-W; 76.62 at SRS)				86	
2,960.5 (770.5 at ANL-W; 2,190 at SRS)				845		1,178.5 (770.5 at ANL-W; 408 at SRS)				924	
100.7 (10.7 at ANL-W; 90 at SRS)				12.8		27.2 (10.7 at ANL-W; 16.5 at SRS)				14.1	
Person-rem	LCF	Person-rem	LCF	Person-rem	LCF	Person-rem	LCF	Person-rem	LCF	Person-rem	LCF
0.03	0.000015	0.0012	6.0 × 10 ⁻⁶	0.14	0.000072	0.03	0.000015	0.0012	6.0 × 10 ⁻⁶	0.2	0.0001
0.004	1.6 × 10 ⁻⁶	0.0012	4.7 × 10 ⁻⁷	0.02	7.9 × 10 ⁻⁶	0.004	1.6 × 10 ⁻⁶	0.0012	4.7 × 10 ⁻⁷	0.027	0.000011
less than 1.0 × 10 ⁻⁶	less than 1.0 × 10 ⁻⁹	less than 3.0 × 10 ⁻⁶	less than 1.5 × 10 ⁻⁹	less than 1.0 × 10 ⁻⁶	less than 1.0 × 10 ⁻⁹	less than 1.0 × 10 ⁻⁶	less than 1.0 × 10 ⁻⁹	less than 3.0 × 10 ⁻⁶	less than 1.5 × 10 ⁻⁹	less than 1.0 × 10 ⁻⁶	less than 1.0 × 10 ⁻⁹

^c Annual dose represents the maximum dose in a single year. Population doses (population and worker population) are in person-rem; individual doses are in millirem. The regulatory dose limit for offsite individuals (public) is 10 millirem per year from air exposures and 100 millirem per year for all pathways. The administrative control limit for an individual worker at a DOE site is 2,000 millirem per year.

^d Includes 142 cubic meters of spent nuclear fuel.

^e Includes 25.2 cubic meters of spent nuclear fuel.

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